"Practical not Perfect"

The Honorable Jeffrey S. Merrifield
Commissioner
U.S. Nuclear Regulatory Commission

at the Nuclear Safety Research Conference
Washington, D.C.

October 21, 2003

Good Morning. It is a pleasure to be here at the Nuclear Safety Research Conference and to have the opportunity to share my views with you today on several of the topics on the agenda. I want to acknowledge the important work of the NRC staff as well as our international colleagues who have traveled great distances to be here.

As a regulator, it seems quite obvious to me that research is an area that cannot be ignored. Without sophisticated and state-of-the-art solutions to our regulatory challenges we could not successfully accomplish our safety mission. To me, there is no doubt that timely, credible, and well documented research is critical to ensuring that our regulatory activities are based on sound science and are focused on the most risk-significant issues. Having said this, it is equally important that our research program remain focused on those areas that are vital to our regulatory mission. Our limited research budget must be targeted toward those areas that will ensure that we are ahead of the technological curve of future reactor and material licensing matters, but it must also provide answers to today's safety, security and environmental concerns. I take every opportunity that I can to remind Ashok Thadani and our research staff to ensure that they are striking the right balance. Our ability to succeed as a regulator depends on the discipline to focus on what is most important. Too much anticipatory research aimed at issues that may never materialize is fiscally unacceptable, and brings with it the risk of diverting research dollars and attention from matters that need to be resolved today.

This morning, I will focus on four significant technical areas where research has played an important role in the past and, in some cases, may have an equally important future role as well. The areas I will briefly address are decommissioning, advanced reactors, fuel performance, and materials degradation.

Decommissioning

We are currently in the most active period of decommissioning ever experienced by the NRC. Considering the fact that the Atoms for Peace program is celebrating its 50th Anniversary this year, this level of activity should not be a surprise. Just a sampling of the sites currently in the decommissioning process includes: 19 power reactors, 15 research reactors, 18 uranium mills, 27 complex sites under the NRC Site Decommissioning Management Plan (SDMP)1, 12 former Atomic Energy Commission licensed sites requiring further decommissioning and dozens of smaller facilities undergoing decommissioning on a routine basis.

Given the broad scope of this program, we need to focus on the lessons learned from this decommissioning process and determine what major policy and technical areas need to be addressed to make the site remediation more efficient and practical. In particular, we need to find methods to efficiently decontaminate facilities and return the property into a productive reuse for society. Research has an appropriate role in this effort, but the research will need to be focused to address specific policy and technical issues and trade-offs will need to be made to accommodate budgetary realities.

Based on our experience to date, we have identified several areas requiring major policy decisions. One example is developing a standard for the release of formally contaminated solid material, sometimes referred to as the clearance standard. It is an understatement to say that this is a controversial and complex matter. It is politically thorny because it ultimately raises the question of whether the solid material will eventually end up in consumer products. Our European counterparts allow a different release standard for natural radiation as opposed to so-called "man-made" radiation, even though from a technical perspective there is virtually no difference. In Europe however, there is virtually no public outcry over the shipment of ore with radiation levels as much as ten times higher than that allowed for the shipment of material with "man-made" radiation. Conversely, the IAEA has developed a one millirem release standard for solid materials that has been adopted by some countries. This is an achievement that we in the United States have yet to accomplish.

I do not agree with those who are tempted to seek further research to determine if we can find a "silver bullet" to resolve what is a straightforward, but tough policy choice. I believe that further research is unwarranted and we must "bite the bullet" and move forward with a realistic practical standard.

Another area where major policy choices are necessary is implementation of the NRC's licensing termination rule. The NRC license termination rule has standards for unrestricted release of a facility or site upon decommissioning as well as standards for decommissioning a site under restricted release conditions. Unfortunately, to date, no site has been able to utilize the NRC's restricted release criteria and the Commission is evaluating how to make restricted release a more viable option.

The Commission is considering a graded approach to institutional controls based on a risk-informed process. On one level, if the risk is low enough, nothing more than a deed restriction may be necessary. As the risk increases, more legally enforceable controls may be required. In some cases, either a government agency assuming long-term custody or perhaps a possession-only NRC license for an indefinite period may be what is needed.

Another problem with institutional controls is the assumption made about future use of the land. The NRC has traditionally used a conservative scenario that assumes the land will be used as a farm with the farming family living on the land and receiving a significant portion of their food grown on-site. This is an overly conservative approach that does not reflect modern realities. The fact is, some land may never be suitable for farming; but productive and safe use of the site may be accomplished if it is restricted to an appropriate industrial or commercial application. This is an approach that has been applied by EPA and many States, and the Commission is considering adopting a similar approach. To fully utilize this important redevelopment option, research efforts will need to focus on improving various calculation techniques to take into account real world land use scenarios and identify practical and realistic outcomes.

What is clear about decommissioning is this simple fact: people who work and live around these sites want them to be useful and productive parts of their communities. Whether the land is used for recreational, industrial, commercial or conservation purposes, stakeholders do not want to have to "worry" about these sites in the future. Our research can and must help us attain this goal.

Advanced Reactors

Now let me make the transition from one side of the spectrum to the other and talk about our efforts on advanced reactors. The political climate has certainly changed in the past few years and nuclear power has gained some prominence in the nation's future energy planning. While there is active work taking place in the House and Senate to conclude the Energy Bill conference, this measure may include incentives for encouraging new reactor orders in the United States. Further, it may also create an experimental nuclear reactor program to produce electricity and hydrogen.

The new reactor designs bring with them many new technical and policy challenges that will have to be addressed by the NRC staff, particularly the Office of Research, and the Commission. To prepare for these challenges, the staff developed an Advanced Reactor Research Plan Infrastructure Assessment. This plan focuses on development of the necessary tools, data and technical bases for supporting an effective advanced reactor licensing process. The scope of the advanced reactor research plan currently includes six advanced reactor designs: Westinghouse's Advanced Pressurized Water Reactor AP1000, General Electric's ESBWR, Atomic Energy of Canada's Advanced CANDU Reactor ACR-700 and General Atomic's Gas Turbine-Modular Helium Reactor (GT-MHR), among others.

Specifically, as many of you know, the NRC staff is currently reviewing the Westinghouse AP1000 design certification application with the final design approval scheduled for September 2004. In addition, pre-application reviews of General Electric's ESBWR, and Atomic Energy of Canada's ACR-700 are targeted for completion on or before mid-2004.

In addition to the design certification and pre-application activities, there are several licensees actively considering their options for new nuclear plant construction in the United States. On September 25, 2003, the staff received the first two Early Site Permit (ESP) applications from Exelon Generating Company for their Clinton site and Dominion Generation for their North Anna site. Similarly, Entergy is expected to submit an ESP application any day now for their Grand Gulf site.

The resurgence of interest in advanced reactor designs and new reactor licensing, also poses very real, near-term resource challenges. Though the new reactor licensing pace is not the same as we experienced in the 1970's, there are new and different challenges before us. While it is important for us to be prepared for the real possibility of a new build and licensing program, any trade-offs will not come at the expense of the current operating fleet. It is imperative that we maintain our focus on the safety of these reactors first, then effectively manage and balance our remaining resource expenditures on new design reviews and orders. Our choices have to be well thought out, focused on those areas that are likely to materialize and those research activities that are absolutely necessary to support our future licensing decisions. I believe a simple, but practical approach is needed. Two areas where our current research efforts will certainly support our future decisions are fuel performance and materials issues.

Fuel Performance

In comparison to where we were 20 years ago, the performance of fuel is greatly improved. The number of light-water-reactor fuel failures has steadily declined during this time. I believe a strong contributor to this improved performance is the increased market competition between the current fuel vendors here in the United States: Framatome-ANP, Westinghouse, and Global Nuclear Fuel/General Electric. Increased competition has forced these companies to review their manufacturing processes and focus on process improvements in the area of new technologies to identify issues such as, chipped fuel pellets and flawed tubes before they are put in service. In addition, vendors are focusing on performance improvements in fuel and cladding design, and other areas to support higher fuel burnups, longer operating cycles, and power uprates.

Yet, despite these successes, the number of fuel failures in the past two years has noticeably increased. Fuel issues are back on the radar screen of many plant operators and calls for improved reliability are common. Thus, the fuel vendors are left with balancing their resources to remain competitive, but still perform the needed research to safely advance their designs.

From where I sit, it appears that industry may be overly focused on the economic issues and may be pushing the fuel too hard. I get concerned when I hear industry folks question whether fuel manufacturers have budgeted sufficient research dollars toward meeting the demand of the new, more aggressive operating environment. From my perspective, increased burnup, longer operating cycles and power uprates are key drivers for the fuel performance desired by our licensees. The fuel environment is going to be more challenging but, as a safety regulator, we need to be assured that the plants can continue to operate safely under these new conditions. To continue to insist on rock bottom fuel prices at the expense of debilitating and costly fuel failures is penny-wise and pound foolish. The industry must leverage its overall experience and utilize initiatives such as the Electric Power Research Institute (EPRI) Robust Fuel Program to effectively deal with fuel reliability.

For our part, the NRC developed a research program to confirm the current fuel burnup limit of 62 giga-watt days per metric ton and to develop a strategy for assessing future requests for burnup extensions beyond the current NRC limit to ensure the adequate protection of public health and safety at our operating reactors. Utilizing a variety of U.S. and international facilities, the NRC research effort is appropriately focused on demonstrating that recent increases in energy output for new cladding alloys can meet our regulatory expectations for postulated accidents. Nonetheless, given the recent spike in fuel failures, I think that both the NRC and industry need to consider additional research to determine how we can get a better handle on new designs and materials that can reverse the recent increase in fuel failures.

Materials Degradation

In a similar way, the NRC has maintained an active research effort for many years that is focused on the management of age-related degradation in nuclear power plants. However, the effectiveness of this program has been recently challenged. Akin to the upswing in fuel failures, materials degradation issues have been pushed to the forefront of the nuclear industry in the past few years. None is more prominent than the reactor pressure vessel head degradation at Davis-Besse. It is nearly 20 months since the discovery of the pineapple-sized cavity in the vessel head at Davis-Besse. What is disturbing to me, is how this was missed.

Our attention was focused on the potential for cracks propagating, turning circumferentially and thus leading to an ejection of a control rod. No one expected the significant erosion of the vessel head itself.

Since this discovery, the NRC and industry have spent a considerable amount of resources reflecting on this event and pondering how it happened. Some in industry believe it is merely one data point and not a reflection of the entire industry. Others question the NRC's oversight and understanding of materials degradation issues, and our ability to effectively manage them. As many of you are aware, the NRC formed a nine-person, lessons-learned task force that spent more than 7000 hours reviewing the NRC's regulatory processes and activities, and provided specific recommendations to the Commission for areas of improvement. Action plans to address these recommendations have been initiated, including recommendations to evaluate plant experience with stress corrosion cracking and boric acid corrosion in order to enhance our inspection requirements and guidelines. In addition, the industry through the EPRI Materials Reliability Project (MRP) is leading the industry's actions to respond to materials degradation issues.

I believe the NRC and industry are very effective at swiftly reacting to these issues when they are discovered. We aggressively search out the root cause and develop action plans to correct identified gaps. However, this reactive approach comes at a cost. When we put ourselves at the mercy of these degradation issues and are forced to react to them, the public loses confidence in the industry and in the NRC as a regulator. In addition, this places significant unanticipated resource strains on our organizations.

As plants continue to mature and more plants pursue license renewal, thereby extending their effective operating life out to 60 years, age-related degradation issues will continue to challenge both the NRC and the industry. These challenges will manifest themselves in new forms of degradation and new locations. Chairman Nils Diaz has said, "we will never have another Davis-Besse." I can't agree more. While we are very good at preventing reoccurrence of issues we have experienced in the past, our research efforts should be focused on identifying emerging and unexpected materials challenges. Like a good physician, we do not expect to merely guess what the future problems of our charges will be. Instead, we need the experience, training and understanding to recognize, diagnose and treat age-related issues before they become critical.

The development of the right tools and methodologies to better predict these age-related degradation mechanisms, so we can become more predictive, rather than reactive, is extremely important. Similarly, like our counterparts in the military, we cannot be fighting our last battle. We must be focused toward our future regulatory challenges, and our research must be well placed to meet that call.

To conclude, it is evident as ever that research plays a critical role in providing the technical advice, tools and information necessary to identify and resolve safety issues. However, our research must be managed in an efficient and effective manner so we can leverage our past experience to further our understanding of potential areas of concern. We must become more proactive in our research efforts and improve our ability to anticipate problems of potential safety significance, thereby limiting the amount of costly reactive work. The resource challenges I mentioned today not only impact us in the United States, but extend beyond our borders. Continuing our collaborative efforts is vital to enhancing our ability to make sound practical decisions based upon our collective worldwide experience.

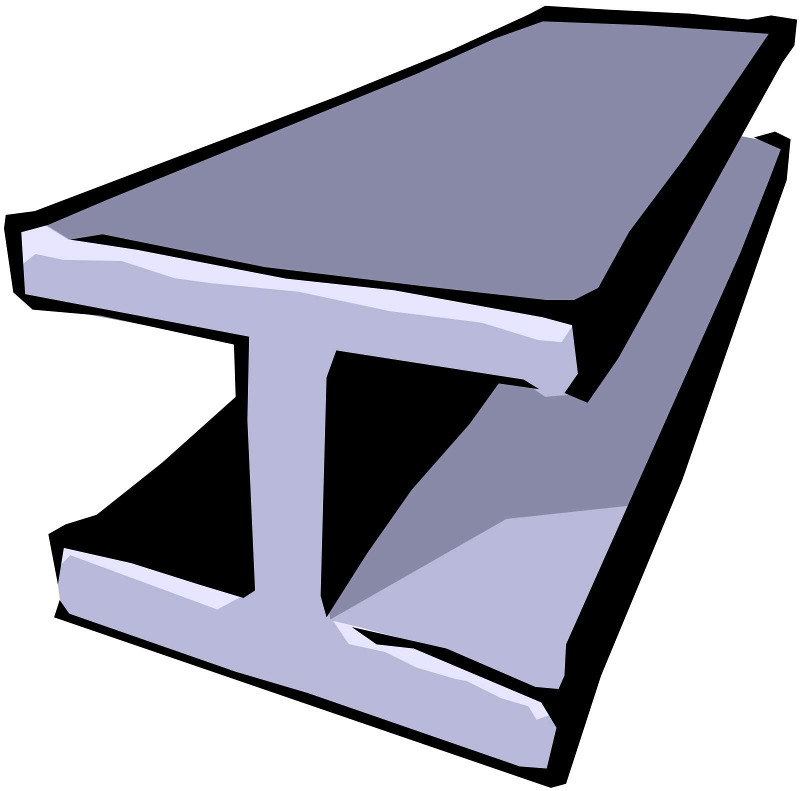
In closing, let me again express my appreciation for the opportunity to share my views with you this morning. I hope you continue to have a successful conference. I would be pleased to answer any questions you may have at this time.

FINALIZATION OF NUREG-1640

Radiological Assessments for Clearance of Equipment and Materials from Nuclear Facilities

Robert A. Meck, NRC
R. Anigstein, H.J. Chmelynski, D.A. Loomis, J.J. Mauro, R.H. Olsher, W.C. Thurber, SC&A
S.F. Marschke, Gemini Consulting Co.

MATERIALS AND EQUIPMENT

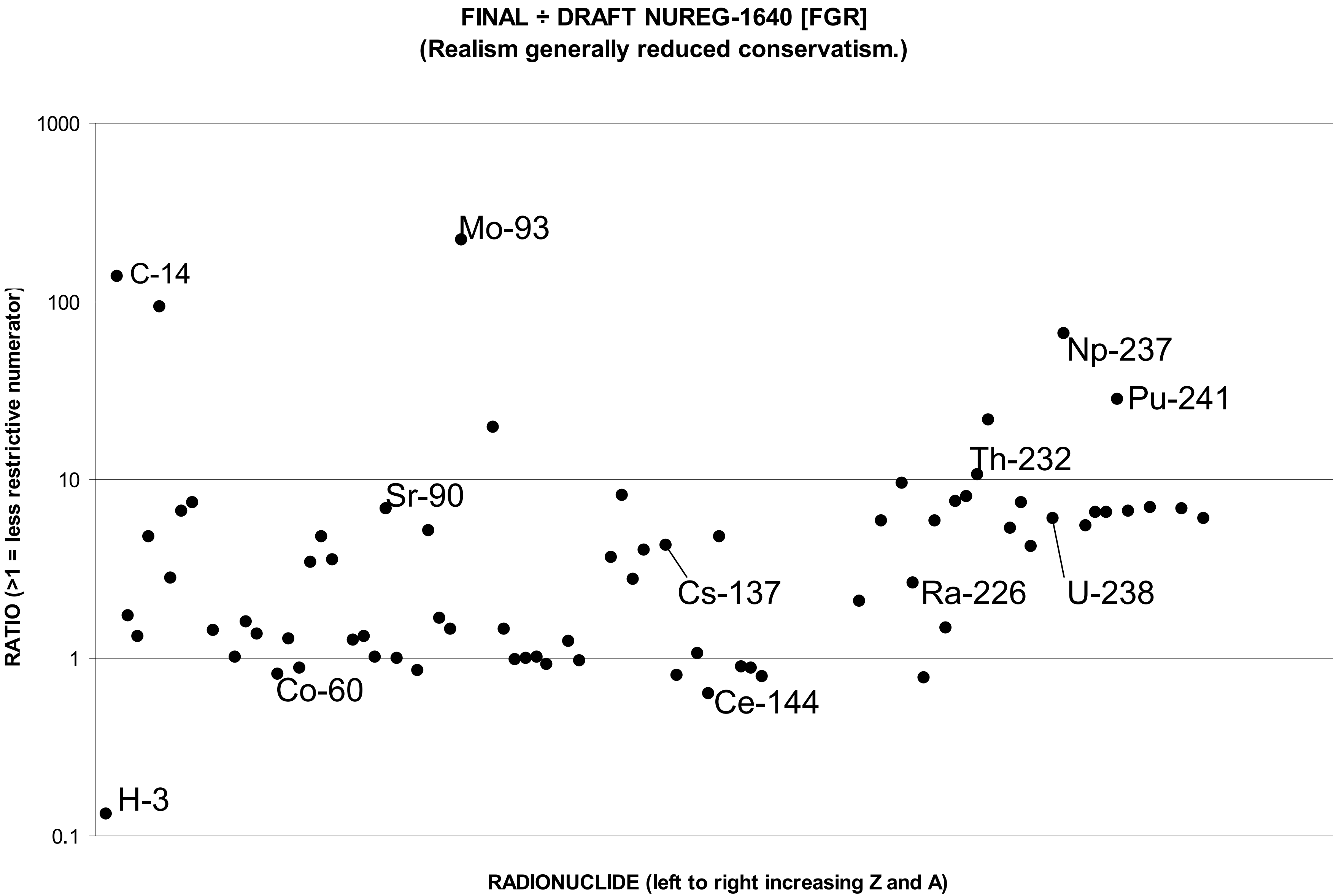


- STEEL
- ALUMINUM
- COPPER
- CONCRETE

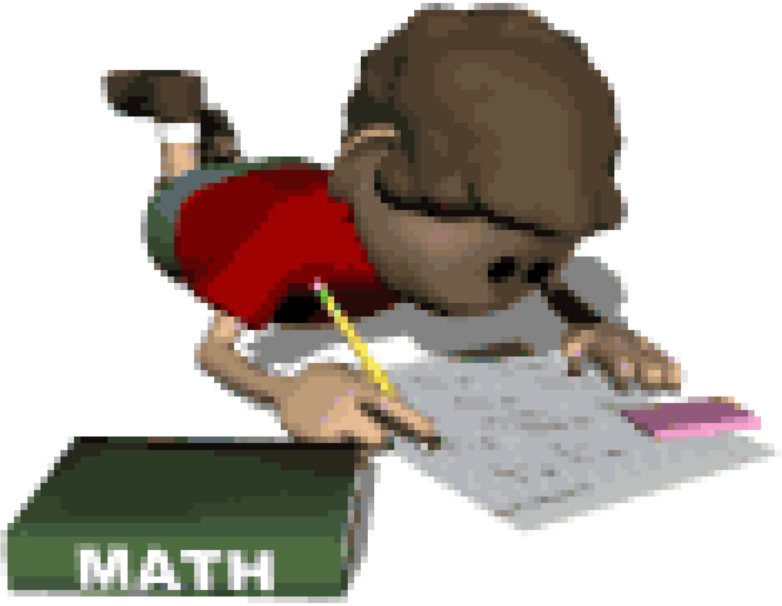
- VOLUMETRIC AND SURFICIAL RESULTS

RESOLVE COMMENTS

- Realism for industry
  - Basic Oxygen, EAF, Induction Furnaces
  - Mixing
  - Partitioning
  - Representations of transportation, copper, aluminum industries
- Less conservatism


FINAL ÷ DRAFT NUREG-1640 [FGR]
(Realism generally reduced conservatism.)
1000
100
10
1
0.1
RATIO (>1 = less restrictive numerator)
C-14
Mo-93
Np-237
Pu-241
Th-232
Sr-90
Cs-137
Ra-226
U-238
Co-60
Ce-144
H-3
RADIONUCLIDE (left to right increasing Z and A)

RESOLVE COMMENTS

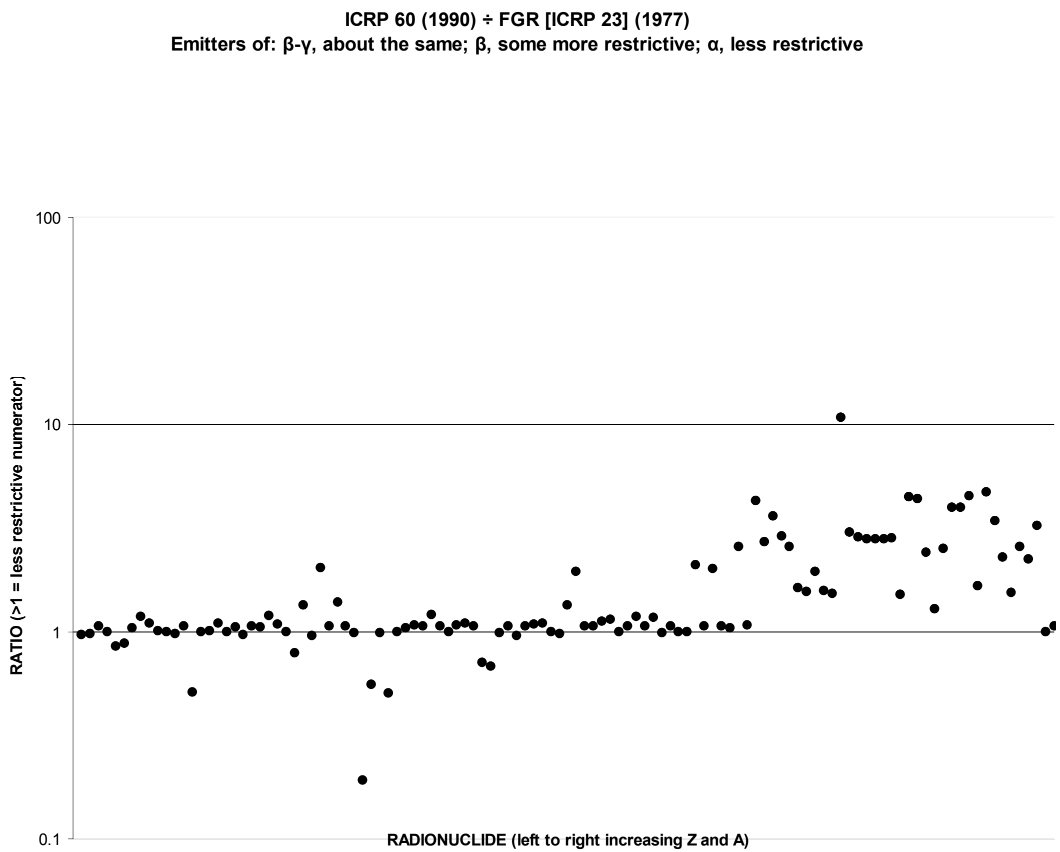


- More accurate geometries
- Drinking water
- More radionuclides
- ICRP 26 AND 60

RESULTS

- ICRP 60 results are mostly the same order of magnitude, but often less restrictive
- ICRP 60 gives generally lower exposures for high Z nuclides—α-emitters
- Mostly job-related critical groups

COMPARISONS



CRITICAL GROUP SCENARIOS--STEEL

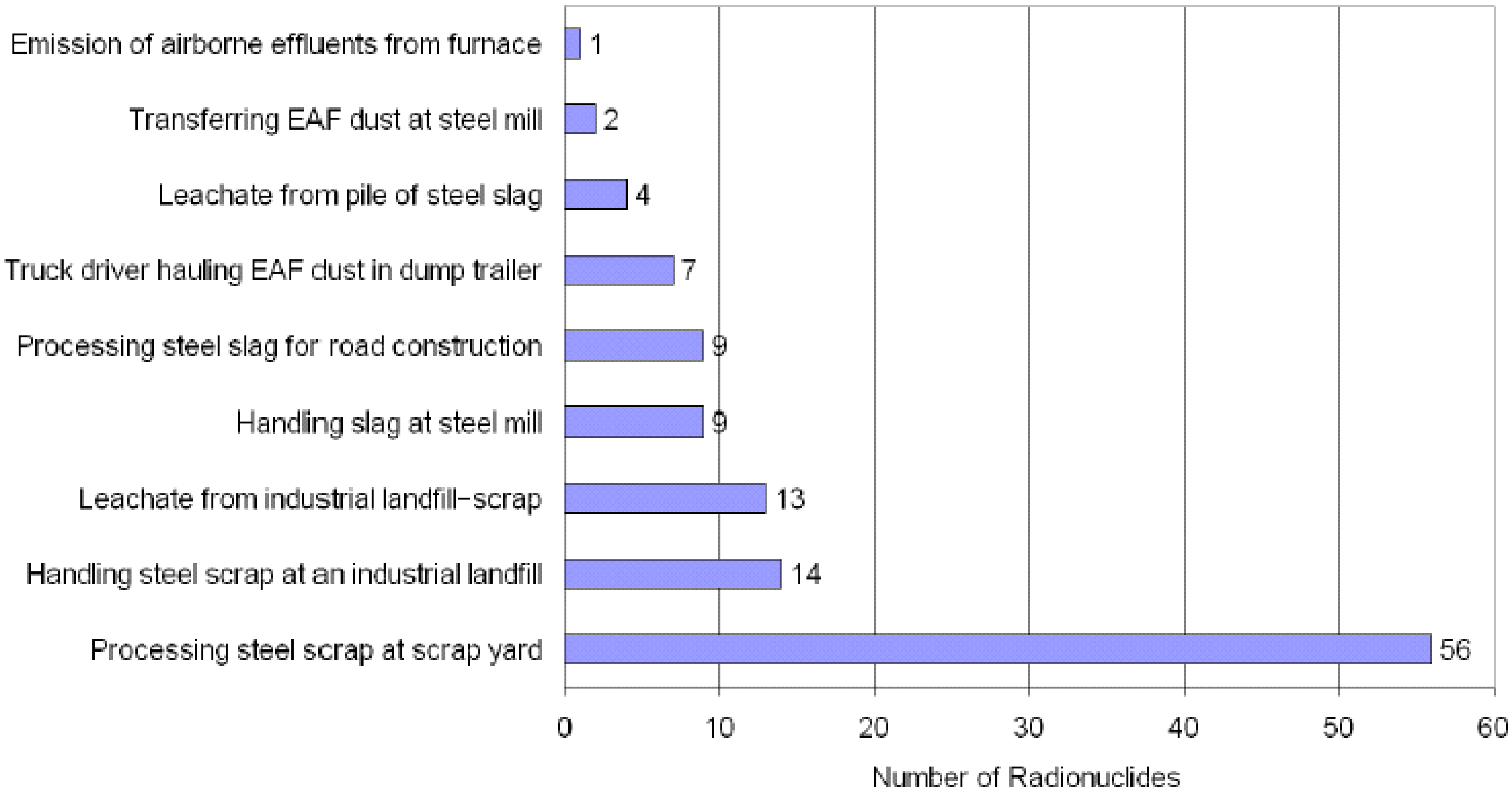


Figure 3.6 Scenarios giving rise to effective dose-critical groups for steel

CRITICAL GROUP SCENARIOS--CONCRETE

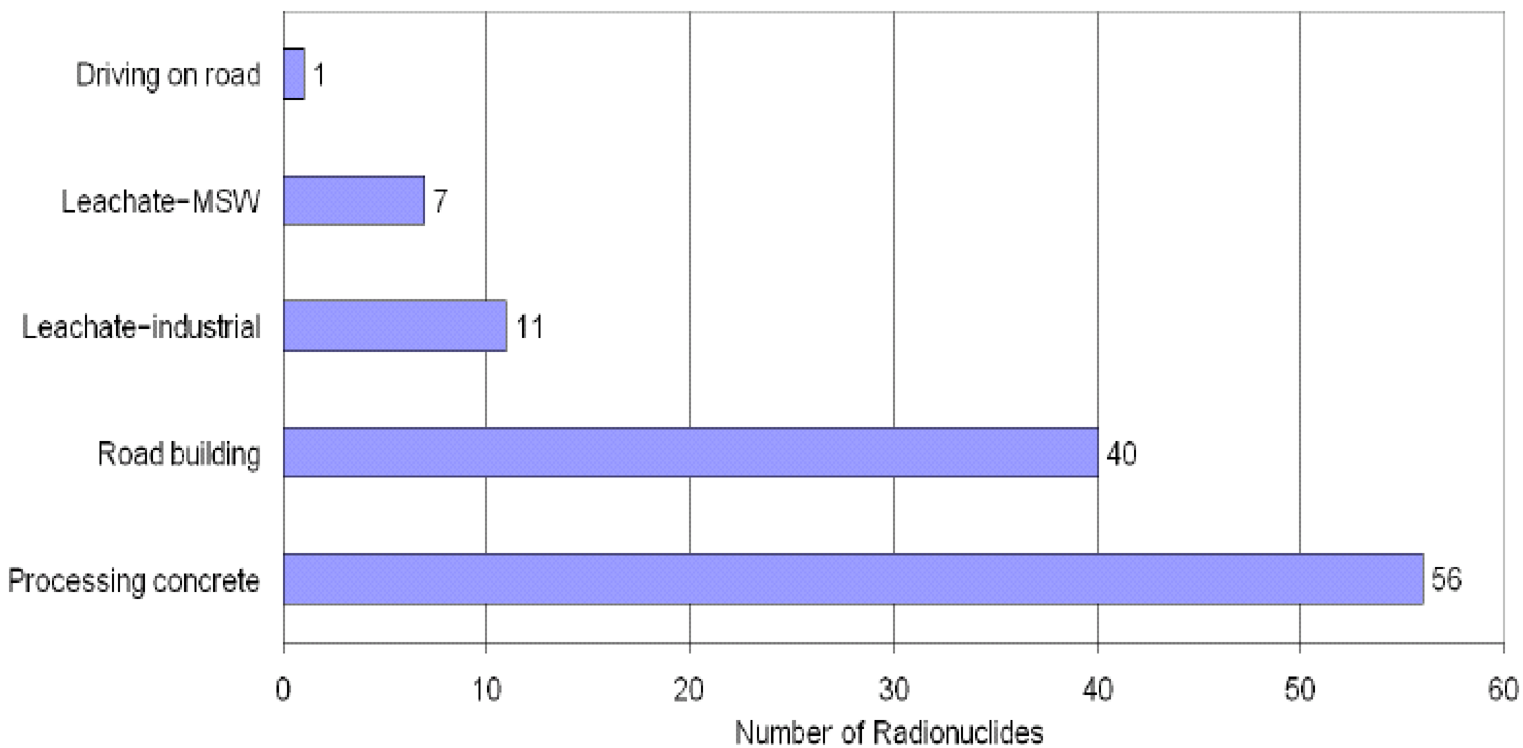
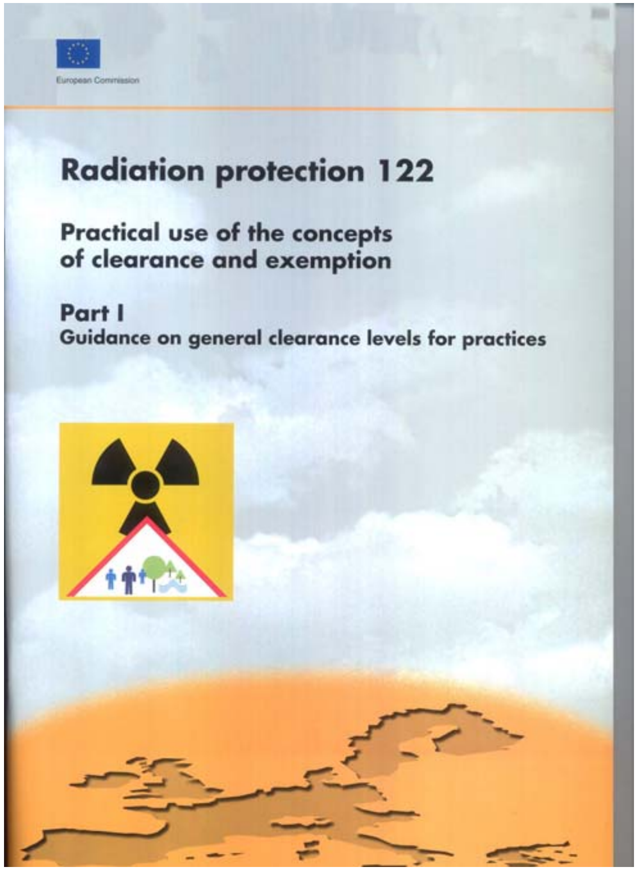


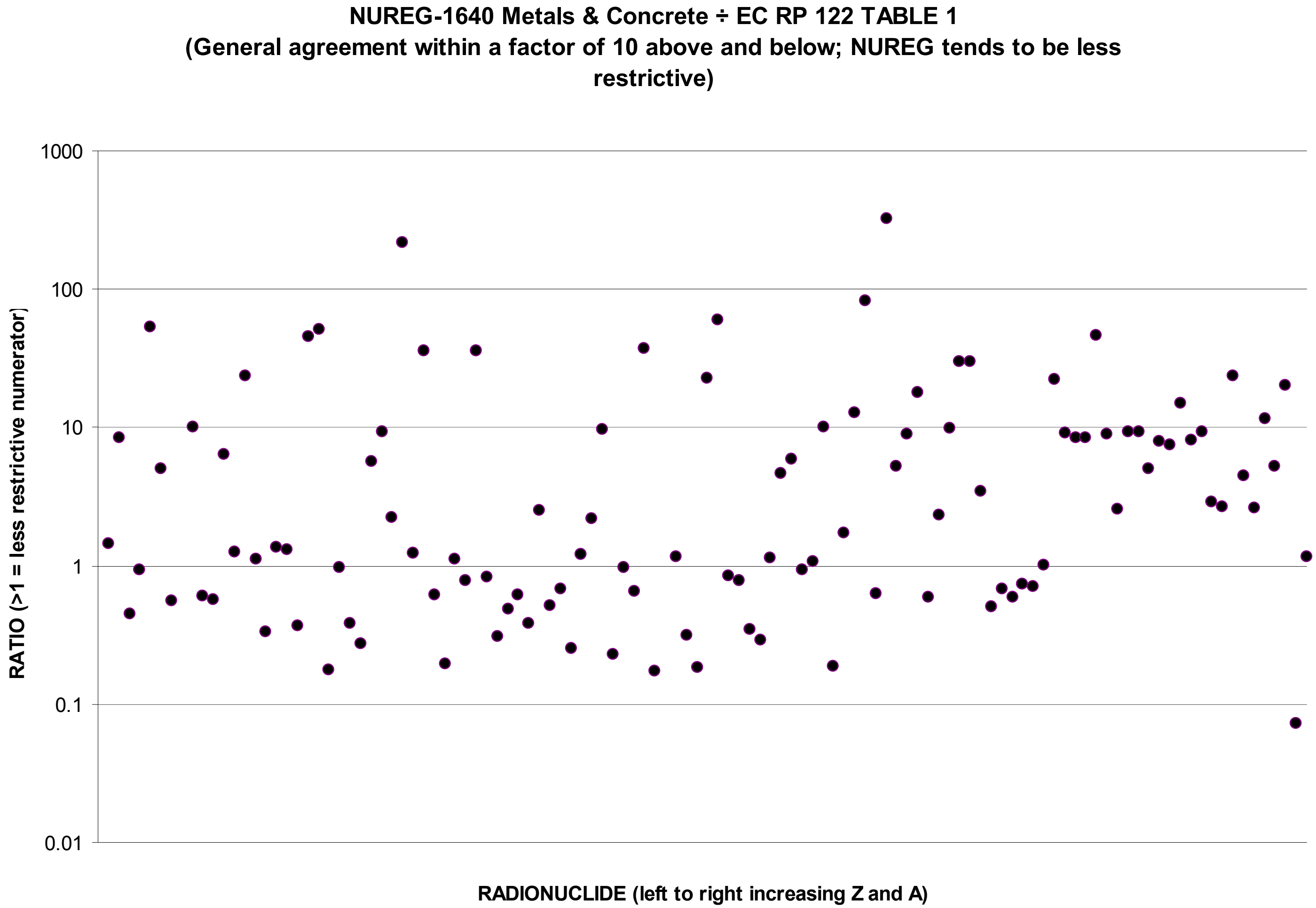
Figure 6.3 Scenarios giving rise to effective dose-critical groups for concrete

COMPARISONS



- NUREG-1640 calculations compare reasonably well with those of the European Commission, considering the very different approaches in making the estimates.

COMPARISONS



PUBLICATION

VOLUMES 1 & 3 ARE PUBLICALLY AVAILABLE

ON THE NRC WEBSITE

VOLUMES 2 & 4 ARE COMING SOON !

Work is in progress!

REUSE OF EQUIPMENT WILL FOLLOW IN A SUPPLEMENT

Multi-agency radiation survey and site investigation manual (MARSSIM) supplements: Overview/development update.

C. Gogolak, C. Petullo, K. Klawiter,
V. Lloyd, R. Meck, H. Peterson,
S. Doremus, D. Caputo, D. Alberth,
G. Powers, A. Huffert, L. Fragoso, R. Bhat

Environmental Measurements Laboratory

201 Varick Street, 5th floor
New York, NY 10014

MARSSIM Background

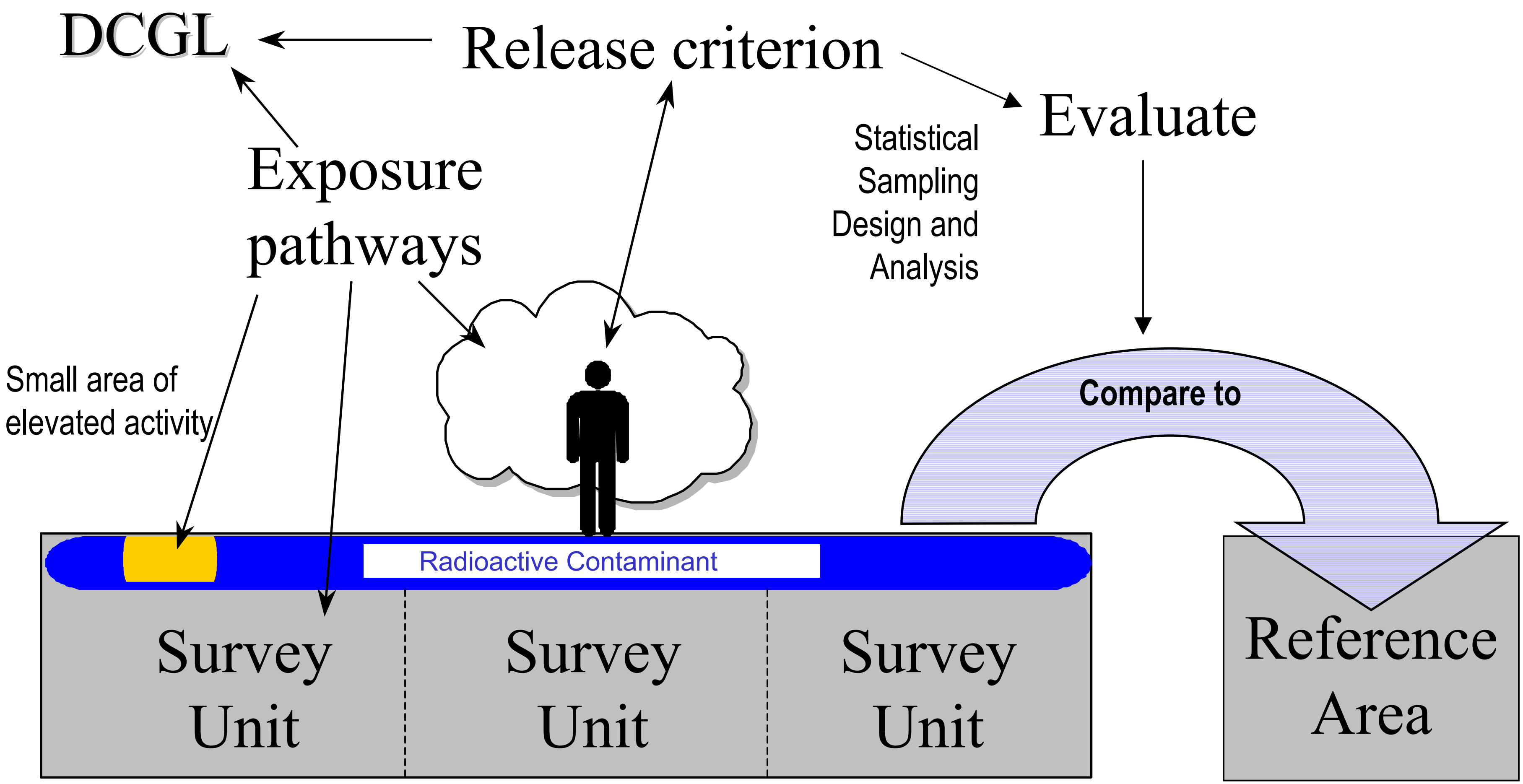
- Thousands of sites with known or potential radioactive contamination
- EPA, NRC, DOD and DOE have, use, or will use dose- and risk-based regulations for site release
- Different agencies use or have used different guidance to demonstrate compliance
- Sites vary - e.g., a single room in a building to large weapons complex sites
- Contamination can be uniform or there can be small areas of elevated radioactivity, even after remediation
- Many site contaminant radionuclides are present in background

MARSSIM Scope

- **Surface soils and building surfaces**
  - How to determine if regulatory release criteria are met
  - How many measurements should be taken
  - What measurement methods to use

- Scope does NOT include:
  - Selecting the release criterion
  - Translating dose or risk into concentrations
  - Groundwater or drinking water compliance
  - Subsurface soil
  - Building materials and release of components
  - Evaluation of remedial alternatives
  - Public involvement

MARSSIM DQO/DQA Process



MARSSIM Timeline

- Developed by DOD, DOE, EPA, NRC
- Effort began in 1994
- Originally released December 1997
- Revision 1 released August 2000
- Additional updates released in June 2001 and August 2002
- Additional updates discussed and approved yearly at April Workgroup Meeting

MARSSIM Workgroup Development of User Tools

- **MARSSIM Webpage:**

http://www.epa.gov/radiation/marssim/

- Frequently Asked Questions (for technical and non-technical personnel)
- Internet Links and Application Software
- Training Dates and Locations
- Document Downloads
- Workgroup Meetings and Contact Information

MARSSIM Workgroup Development of User Tools

- **Frequently Asked Questions (General)**
  - What is MARSSIM?
  - How does MARSSIM affect me?
  - How does MARSSIM fit into existing programs?
  - How does MARSSIM work?
- **Frequently Asked Questions (Technical)**
  - The DCGL (Derived Concentration Guidance Level)
  - The MDC (Minimum Detectable Concentration)
  - The LBGR (Lower Bound of the Grey Region)
  - Double Sampling
  - Size of the Survey Unit

MARSSIM WG News

- **Department of Homeland Security invited to join the MARSSIM Workgroup**
  - Accepted in September 2003
  - New charter and logo under development

MARSSIM Workgroup Development of Manual Supplements

- **Two supplements under development:**
  - Multi-Agency Radiation Survey and Assessment of Materials and Equipment (MARSAME)
  - Multi-Agency Radiation Survey and Assessment of the Subsurface (MARSAS)

MARSAME
(Multi-Agency Radiation Survey and Assessment of Materials and Equipment)

PURPOSE:

To provide consistent measurement guidance for release of materials and equipment, based on the DQO Process, to optimize a release survey protocol based on:

- Material being released
- Available survey instrumentation
- Applicable release criteria
- Location of radionuclides, i.e., surfaces or volumes

MARSAME Scope

- **Materials**
  - scrap metals
  - concrete rubble
  - disbursable bulk materials
  - building debris and excavated soils Bulk soils
- **Equipment**
  - reuse of equipment; hand-held or large

[i.e., Non-real property]

Relationship to MARSSIM

- **MARSAME is a MARSSIM *Supplement***
  - Uses MARSSIM framework for survey design and assessment
  - Extends DQO process to issues specific to surveys of materials and equipment
  - Focuses on demonstration of compliance with release criteria
- ***Unlike* MARSSIM, MARSAME users can decide whether or not to survey**
  - Real property must be surveyed, for both conditional and unconditional release
  - Materials and equipment may be disposed of as radioactive material instead of surveyed for release
  - Survey feasibility and cost may be determining factors in the decision

MARSAME Survey Design

- **Considerations include:**
  - Physical description or condition
  - Difficult-to-access areas
  - Potential for and nature of radioactivity
  - Process knowledge/historical data
  - Release criteria
  - Limitations of field measurement technology
  - Need for sampling and laboratory analysis

MARSAME Survey Design

- ***Results must demonstrate release criteria have been met with acceptable controls on decision error.***
- ***Survey unit size and areal/volumetric averaging must be developed with a technically defensible approach.***
  - Volumetric Residual Radioactivity – Residual radioactivity residing in or throughout the volume of a solid, liquid, or gas
  - Inaccessible areas. No area is inaccessible, it is a matter of the effort required to access the area. MARSAME term is "difficult-to-access areas."

MARSAME Survey Measurement Methods

- Scanning-only surveys
  - Conveyorized scanning
  - Hand-held meter scanning
- *In toto* surveys
  - Box, tool, or drum counters
  - Portal monitors
- Direct measurements
  - *In situ* gamma spec
  - Hand-held meter
- MARSSIM-type survey
  - Combination of statistical sampling and analysis and scanning surveys

Survey Unit Coverage

- **Recommended coverage based on classification**
  - Class 1 – 100%
  - Class 2 and 3 – Graded percentage
- **Methods for obtaining recommended coverage**
  - Scanning
  - *In toto* measurements
  - Direct measurements

For many surveys, these methods may be sufficient for decision making, provided the chosen method has adequate sensitivity relative to the release criteria.

Role of Statistical Sampling

If the chosen measurement method can achieve the recommended survey unit coverage with adequate sensitivity then statistical sampling is may not be necessary:

If scan MDC **<** $DCGL_{ME}$, 100% Scan for Class I material with all measurements less than the $DCGL_{ME}$ should be sufficient to make a decision that the release criteria have been met with acceptable limits on decision error.

If scan MDC is ≥ $DCGL_{ME}$ then statistical sampling as described in MARSSIM will probably be necessary.

$DCGL_{ME}$ = Derived Concentration Guideline Level for Materials and Equipment

Some MARSAME Survey Measurement Method Particulars (preliminary)

- **Reference area may be the object itself**
  For example: a pre-operational survey of the object is performed before entry into an impacted area. This baseline background information is used as the reference for the release survey.
- **Closure versus operational releases (or restricted versus unrestricted releases)**
  Survey DQOs will not be different
- **Smears**
  Ref. MARSSIM Page 8-25 “used as a diagnostic tool to determine if further investigation is necessary
- **Sentinel measurements can be used to determine impacted versus non-impacted areas or volumes**
  - Definition and scope under development
- **Class 3 example**
  - Expected to account for majority of release scenarios
  - Difficult-to-access areas addressed
  - Portions under development
- **Comprehensive flow chart under development**

Some Answers to FAQs

MARSAME will address the interface with dose modeling at about the same level of detail as MARSSIM

MARSAME will include more information on data quality to ensure accuracy of measurements. A Section, appendix, and FAQs are currently under development.

MARSAME will address special issues of accelerator materials, orphan sources and TENORM within it's overall approach.

MARSAME will address the problem of alpha contamination from background Rn and progeny when surveying.

Potential example scenarios:

- University or R&D laboratory
- Accelerator
- Pile of demolition debris
- Nuclear power plant
- Mine/mill property
- Depleted uranium scenario
- Hospital facility (e.g., trash)

Status and Timeline

- **MARSAME Draft 5 under development (10 drafts to finalize MARSSIM)**
- **MARSSIM Workgroup reviewed Draft 5 in August 2003**
- **Additional drafts and workgroup meetings planned for fall and winter 2003**
- **EPA's Science Advisory Board – Radiation Advisory Committee**
  - Initial Consultation held February 2003
  - Additional Consultation October 2003 -TODAY!
- **Internal Agency Reviews expected mid-2004**

Multi-Agency Radiation Survey and Assessment of the Subsurface (MARSAS)

- **PURPOSE:**
  To develop Final Status Survey design methods and protocols that parallel MARSSIM for performing final status surveys in subsurface soils.

Need for MARSAS

Defining the Problem:

- Some sites have subsurface radioactivity that can't be measured from the surface.
- Need to protect groundwater
- Need to find dispersed and discrete sources

Identifying the Decision:

- Does the volume of subsurface soil meet the release criterion

Subsurface Compartments

- Unsaturated Zone (Vadose) **– occurs above the water table. Pores in the soil, sediment, or rock are only partially filled with water. Measurement focuses on the solids in the soil.**
- Ground Water **- occurs below the water table where pores are completely filled with water. Measurement focuses on the water.**

MARSAS Scope

- Currently, scope limited to the design and performance of “a snapshot in time”, final status surveys for free and/or restricted site release.

- Out of Current MARSAS Scope
- Monitoring integrity of remedial actions (e.g., landfill caps or waste stabilized in place)
- Monitoring over time to assess changes in conditions at a site
- Groundwater

MARSAS Framework: Questions Before the Workgroup

- How do you make the decision that the subsurface is non-impacted?
- How do you determine that remediation is necessary?
- How do you defend/justify leaving known or suspected subsurface radioactive material in-place?

Active Research Areas

- Better Survey Design
- Better Data Analysis
- Cannot scan 100% in Class 1 – How do we keep confidence High and uncertainty Low?
- Sampling Costs – need mechanized equipment (e.g., drill rig, push-probe rig)
- Analytical Costs – third dimension adds additional layers to the decision, each layer requires sampling
- Vadose zone – Groundwater interface
- Dispersed groundwater plume versus discrete sources (e.g., "buried treasure")
- Incorporating surrogate data into the decision process (e.g., geophysical)

MARSAS Status and Plans

- **MARSAS in DRAFT 1**
  - MARSSIM took approximately 10 Drafts to finalize
- **EPA-SAB-RAC Consult conducted February 2003**
  - Are there any comments on our technical approach?
  - Is there anything missing in our technical approach?
  - Any suggestions for additional approaches to this problem?
  - What pitfalls should we be aware of?

Scope may be altered to address comments received

- **WG is reviewing EPA-SAB-RAC & Literature Search results prior to further developing chapters**
- **Resources focused on MARSAME development until internal agency reviews underway**
- **Internal Agency Review – First Qtr. FY-05 (tentative)**

Additional Information

Visit the MARSSIM website at:

http://www.epa.gov/radiation/marssim/

Modeling Tools to Support Decision-Making in Site Cleanup and Decommissioning

Advancements in the RESRAD Family of Codes: RESRAD-OFFSITE and RESRAD-BIOTA

Nuclear Safety Research Conference
October 21, 2003

S. Domotor (DOE), T. Mo (NRC),

A. Williams (DOE) and C. Yu (ANL)

U.S. Department of Energy

Office of Air, Water and Radiation Protection Policy & Guidance

Nuclear Safety
Research Conference

Technical Session:

Advanced Reactors

Session Objective

Present and discuss planned or ongoing research that is needed to form a sound technical basis for safety, regulatory or licensing decisions on advanced reactors.

Advanced Reactor Topics

- NRC Safety Research Program (NRC)
- DOE Research at the National Labs (INEEL)
- Analytical Codes and Data Needs (NRC)
- New Plant Licensing Framework (NRC)
- Gen IV Concepts and Challenges (DOE)

Nuclear Safety Research Conference

Advanced Reactor Research Plan

John H. Flack, Branch Chief

Regulatory Effectiveness and Human Factors Branch

Office of Nuclear Regulatory Research

jhf@nrc.gov

October 21, 2003

Outline

- Commission advanced reactor policy
- Key policy issues and framework
- Early identification of technical and safety issues
- Development of necessary technical infrastructure
- Challenging technical and safety issues
- Regulatory challenges
- Summary

Commission Advanced Reactor Policy

- Innovative reactor designs are encouraged
- Increased safety margins are expected
- Innovative licensing criteria can be proposed
- Designers must address defense-in-depth, safety goal and severe accident policies, industry codes and LWR regulations
- License requires sound basis for new technology
- Prototype testing is encouraged
- Early interactions with NRC are encouraged

Consistency with Advanced Reactor Policy

- Less Complexity – Passive Features
- Longer Time Constants
- Reduced Operator Action
- Minimized Potential for Severe Accidents
- Reliable Balance-of-Plant
- Easily Maintainable
- Multiple Barriers

Advanced Reactor Policy Issues (Non-LWRs)

- Evaluation of improved “safety margins”
- Achievement of adequate “defense-in-depth”
- Use of international codes and standards
- Use of PRA in event selection, equipment classification, in place of single failure criteria
- Mechanistic source term acceptance basis
- Confinement vs containment acceptance basis
- Reduced emergency planning acceptance basis

New Regulatory Framework

- Existing framework is based on current LWRs
- Advanced reactors use different technologies and approach to defense-in-depth
- New framework features:

  - Technology neutral
  - Utilize PRA results and insights
  - Less prescriptive performance based

Regulatory Research Infrastructure Development Areas

- Research facilities
- Staff knowledge, skills and abilities
- Independent analytical codes and methods
- Experimental data
- Contractor support

Regulatory Research Infrastructure Development Issues

- Adequacy of Industry R&D
- Independent confirmatory capability
- Assessment of beyond the design basis
  - Probe failure conditions; identify margins (severe accident knowledge)
  - Key areas (e.g., fuel integrity)
- Reliance on international cooperation

Infrastructure Development

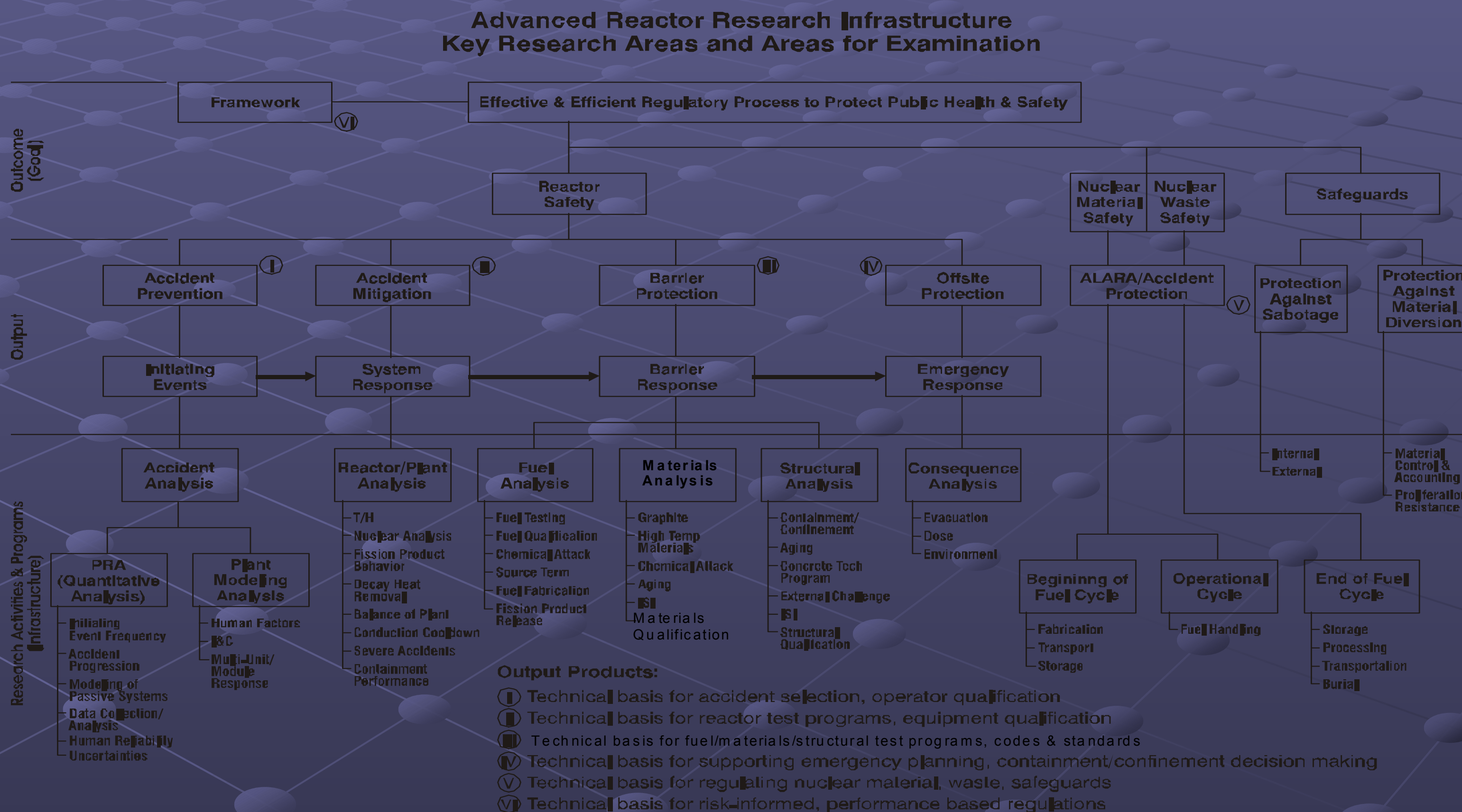


Figure 1 Key Research Areas for Examination

Advanced Reactor Technical Challenges

- PRA models, data, uncertainties, metrics
- Advancements in Instrumentation and control
- Staffing for passive and multi-module plants
- Fuel as primary fission product barrier
- High temperature materials performance
- Thermal-hydraulic & severe accident analysis

Summary of NRC Advanced Reactor Research

- Identify key policy and safety issues
- Develop technical review infrastructure
- Support resolution of policy issues
- Develop a new framework
- SECY-03-0047 (Policy Issues)
- SECY-03-0059 (Infrastructure)

Idaho National Engineering and Environmental Laboratory

<u>Ed. Note</u>: This presentation contained numerous high-resolution photographs and drawings and would not have uploaded easily to this site. With the viewer in mind, we are providing only the title page. To request the presentation in its entirety, please contact the authors.

DOE Advanced Reactor Research

James A. Lake, PhD
Associate Laboratory Director
http://nuclear.inel.gov/

Richard R. Schultz, PE
Generation IV Technical Integration Lead

Nuclear Safety Research Conference
Washington DC

October 21, 2003

STATUS OF THERMAL-HYDRAULIC RESEARCH ACTIVITIES RELATED TO ADVANCED REACTORS

Stephen M. Bajorek, Ph. D.
Office of Nuclear Regulatory Research
United States Nuclear Regulatory Commission

Nuclear Safety Research Conference
Washington DC
October 22, 2003

INTRODUCTION

- Since 2000 there has been a re-emergence in design certification activity for advanced water reactors
- Several designs under various stages of review:

Design	Applicant	Type
AP1000	**Westinghouse**	**Advanced Passive PWR**
ESBWR	**GE**	**Advanced Passive BWR**
SWR-1000	**Framatome ANP**	**Advanced Passive BWR**
ACR-700	**AECL**	**Advanced LW Cooled/HW Moderated**
IRIS	**Westinghouse**	**Advanced Passive PWR**
PBMR	**Eskom**	**Advanced Gas Reactor**
GT-MHR	**General Atomics**	**Advanced Gas Reactor**

Objectives / Outline

- Summarize the design features and thermal-hydraulic issues related to advanced reactors are likely to require additional research and development.
- Provide a status on relevant research activities being conducted by Office of Research.

Why Are There Thermal-Hydraulic Issues ?

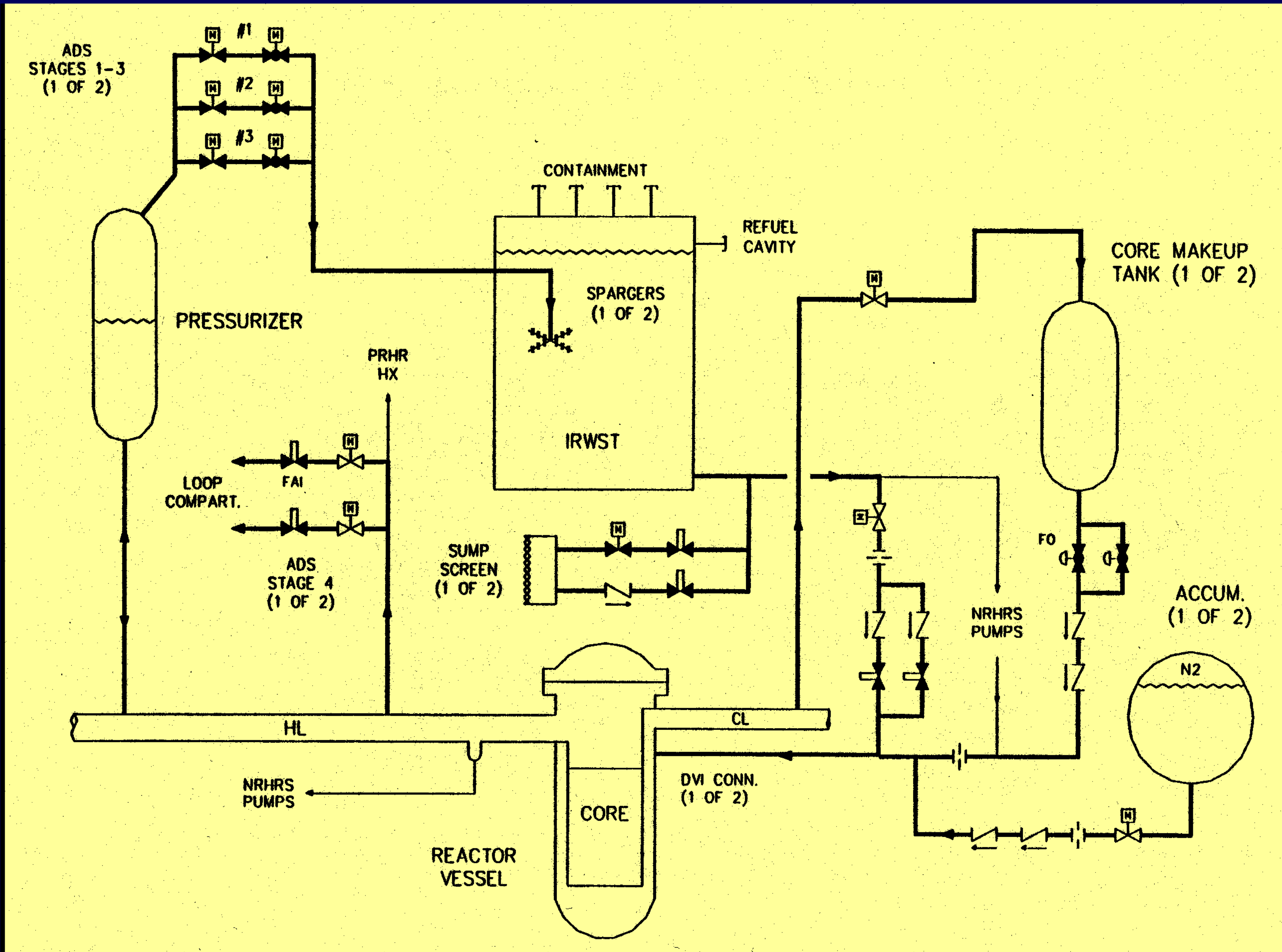
- Passive safety features result in transients dominated by natural circulation and flows driven by small driving heads.
- Some “traditional” accident scenarios eliminated by design. Most limiting accident may not be clearly identifiable.
- New plant components and design features.
- State-of-the-art in boiling, condensation and two-phase flow.

AP1000

AP1000 Passive Safety Systems

- PRHR : Passive residual heat removal system to remove decay heat.
- ADS : Automatic depressurization system to quickly reduce RCS pressure.
- CMT : Core makeup tanks to provide coolant at high pressures.
- IRWST : In-containment refueling water water storage tank to provide large quantity of water to RCS and act as heat sink.

AP1000 Passive Safety Systems

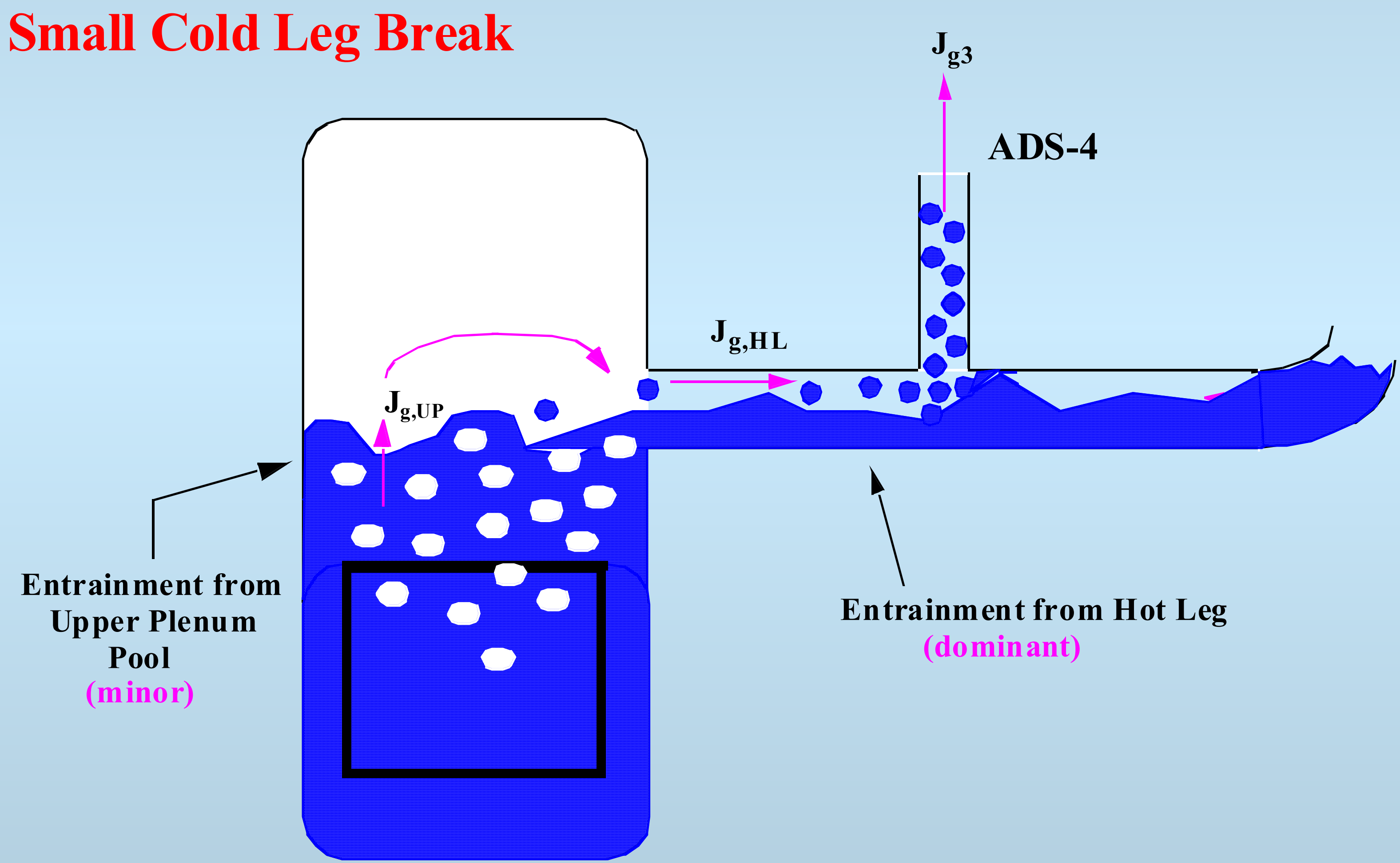


Review Status

- Draft Safety Evaluation Report (DSER) was issued with 174 open issues.   A significant number have been closed since.
- Important remaining thermal-hydraulic issues involve performance of ADS-4 system during ADS-4 blowdown and long-term cooling.

AP1000 Thermal-Hydraulic Issues

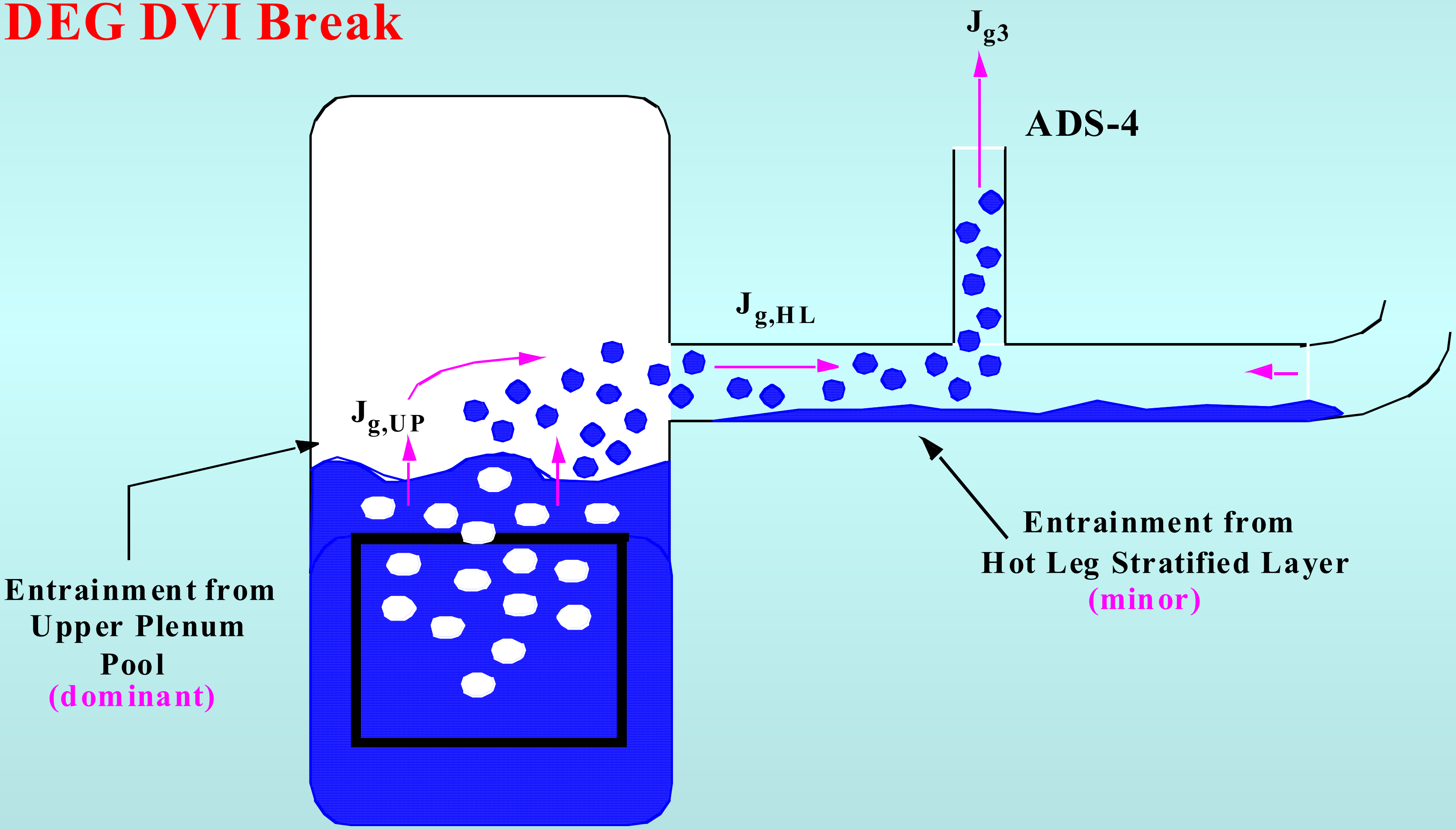
- Hot Leg Flow Pattern & Offtake to ADS-4



AP1000 Thermal-Hydraulic Issues

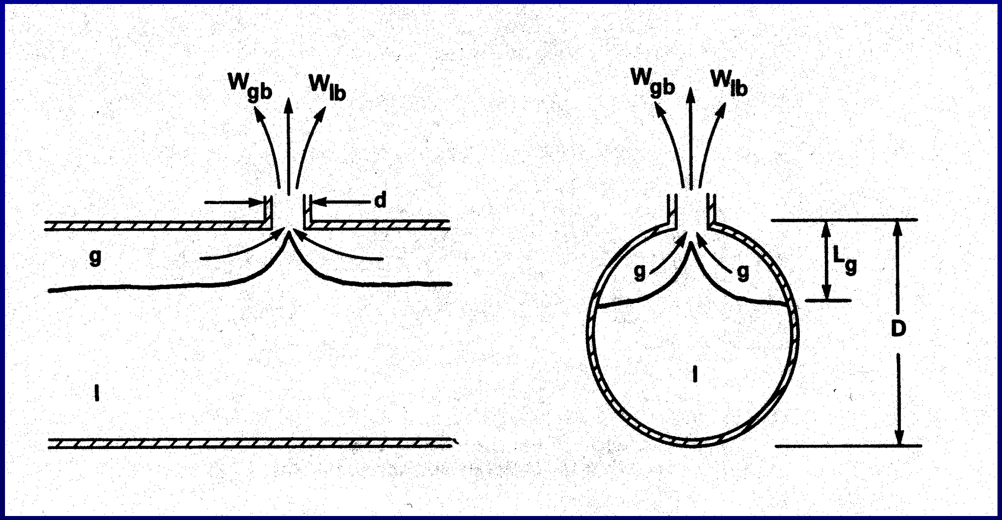
- Upper Plenum "Pool" Entrainment

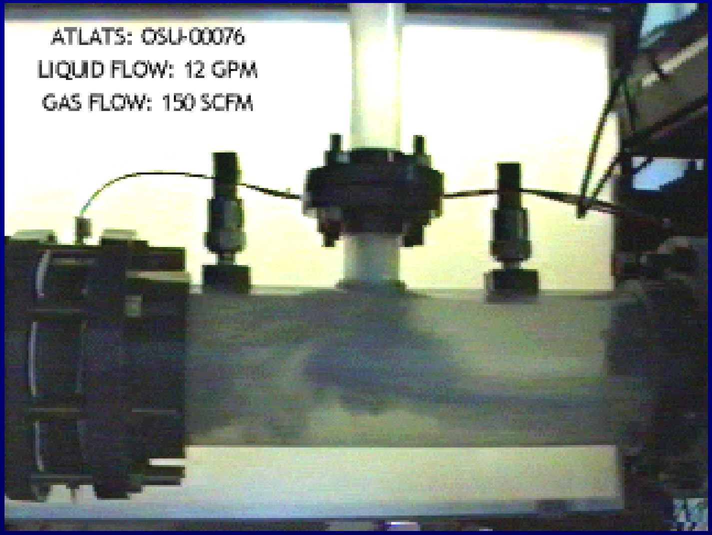
DEG DVI Break



NRC Experimental Investigations

- ATLATS: The ATLATS facility at Oregon State University (OSU) is being used to develop a database and improved models for hot leg entrainment and offtake at an upward facing branch line. Plans have been made to extend investigation to upper plenum entrainment.





Conventional View

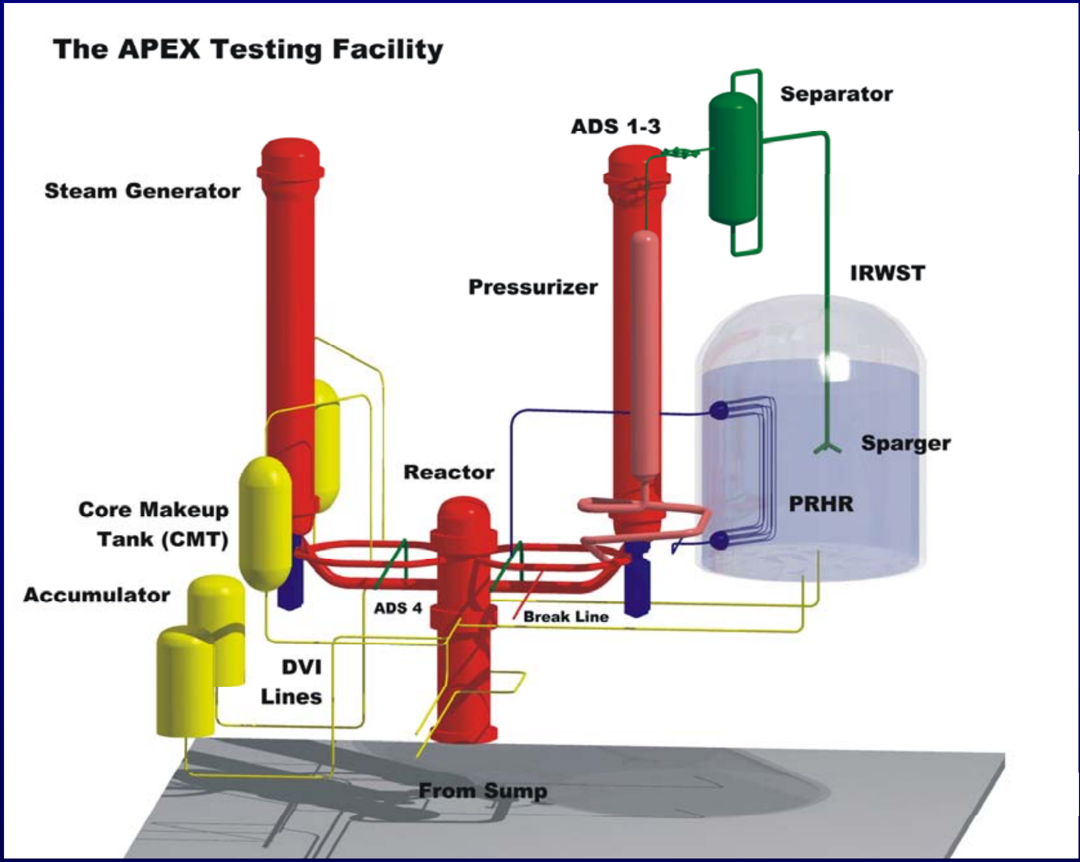
ATLATS Test

NRC Experimental Investigations

- **APEX: Integral tests are planned for the APEX facility, also at OSU, to investigate AP1000 response. APEX is a scaled facility representing the AP600 / AP1000 configuration.**

- **DOE-NERI modifications to APEX include higher core power, larger CMTs and PZR, larger diameter DVI and ADS piping.**

- **NRC sponsored tests include UP "pool" entrainment tests for model development, and integral tests at beyond design basis conditions.**



NRC Sponsored APEX Tests

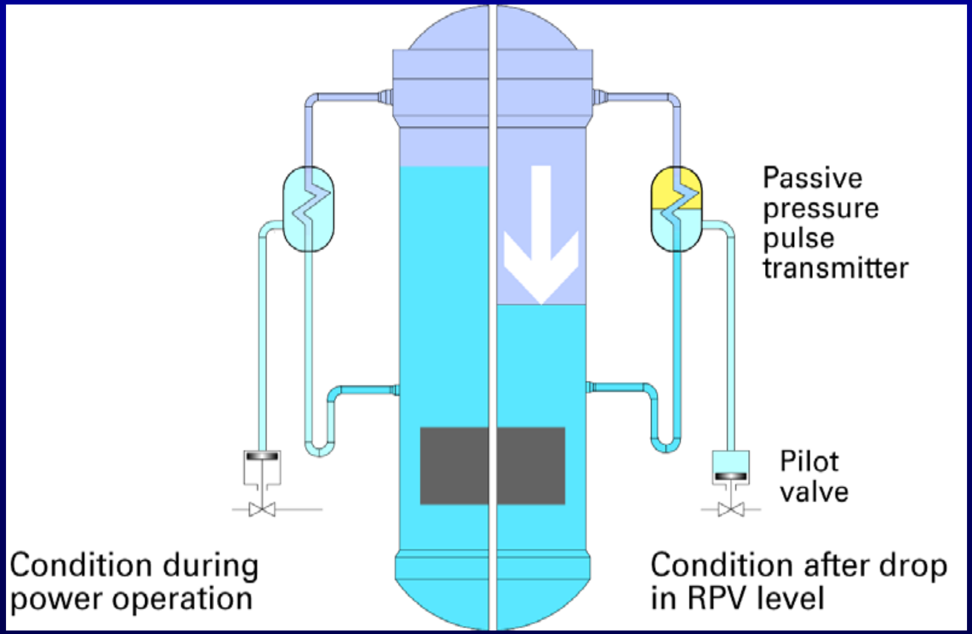
- NRC-AP1000-01: Double-ended DVI Break with Failure of ADS-1/2/3
- NRC-AP1000-03: Double-ended DVI Break with Failure of 2/4 ADS-4 (PZR Side)
- NRC-AP1000-04: 2-Inch CL Break with Degraded Passive Safety Systems
- NRC-AP1000-05: Double-ended DVI Break with Failure of 2/4 ADS-4 (non-PZR Side)
- NRC-AP1000-06: 2-Inch CL Break with Failure of 2/4 ADS-4 (PZR Side)

APEX tests provide independent verification of applicant's claims on safety margin, and provide unique data to assist in evaluation of entrainment processes.

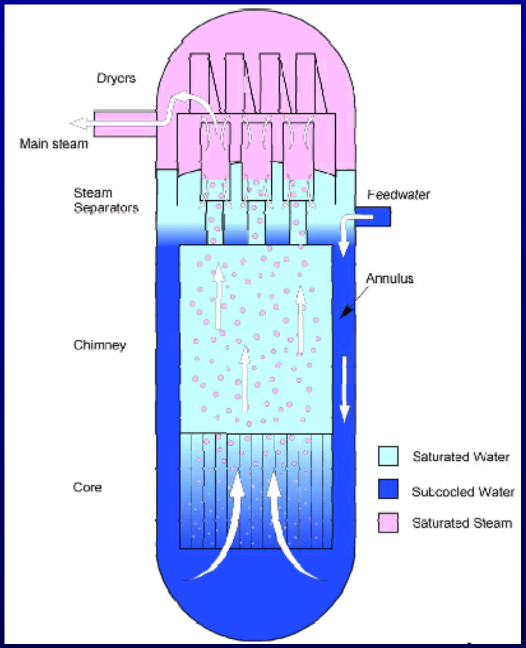
ESBWR / SWR-1000

ESBWR / SWR-1000 Summary

- Both are relatively large capacity units; 4000 MWt /1390 MWe for ESBWR, 1253 MWe for SWR-1000
- No recirculation pumps – reliance on natural circulation.
- Passive safety systems for decay heat removal.

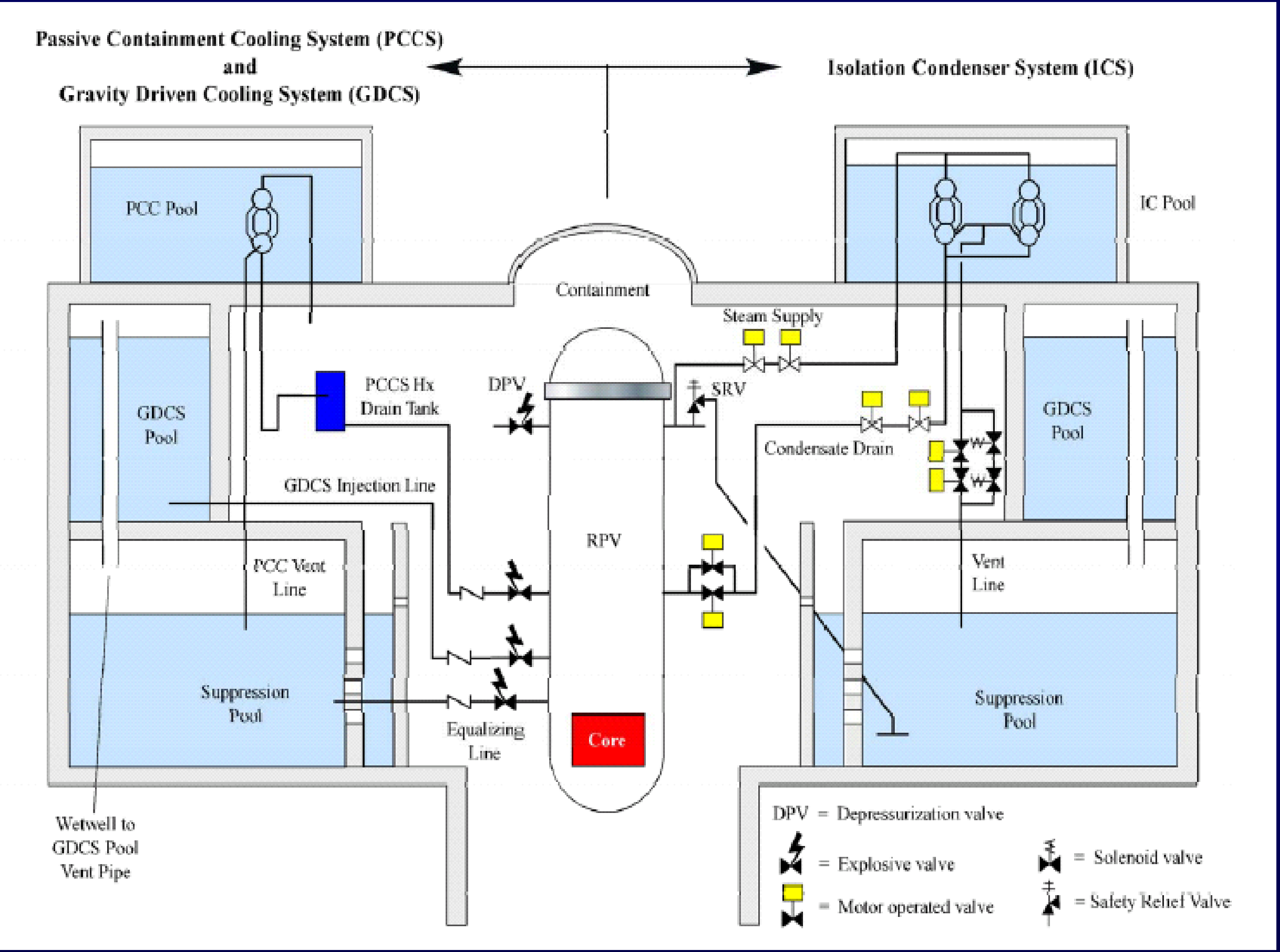


SWR-1000



ESBWR

ESBWR Passive Safety Systems



SWR-1000 Passive Safety Concepts

Containment cooling condensers for containment heat removal (4)

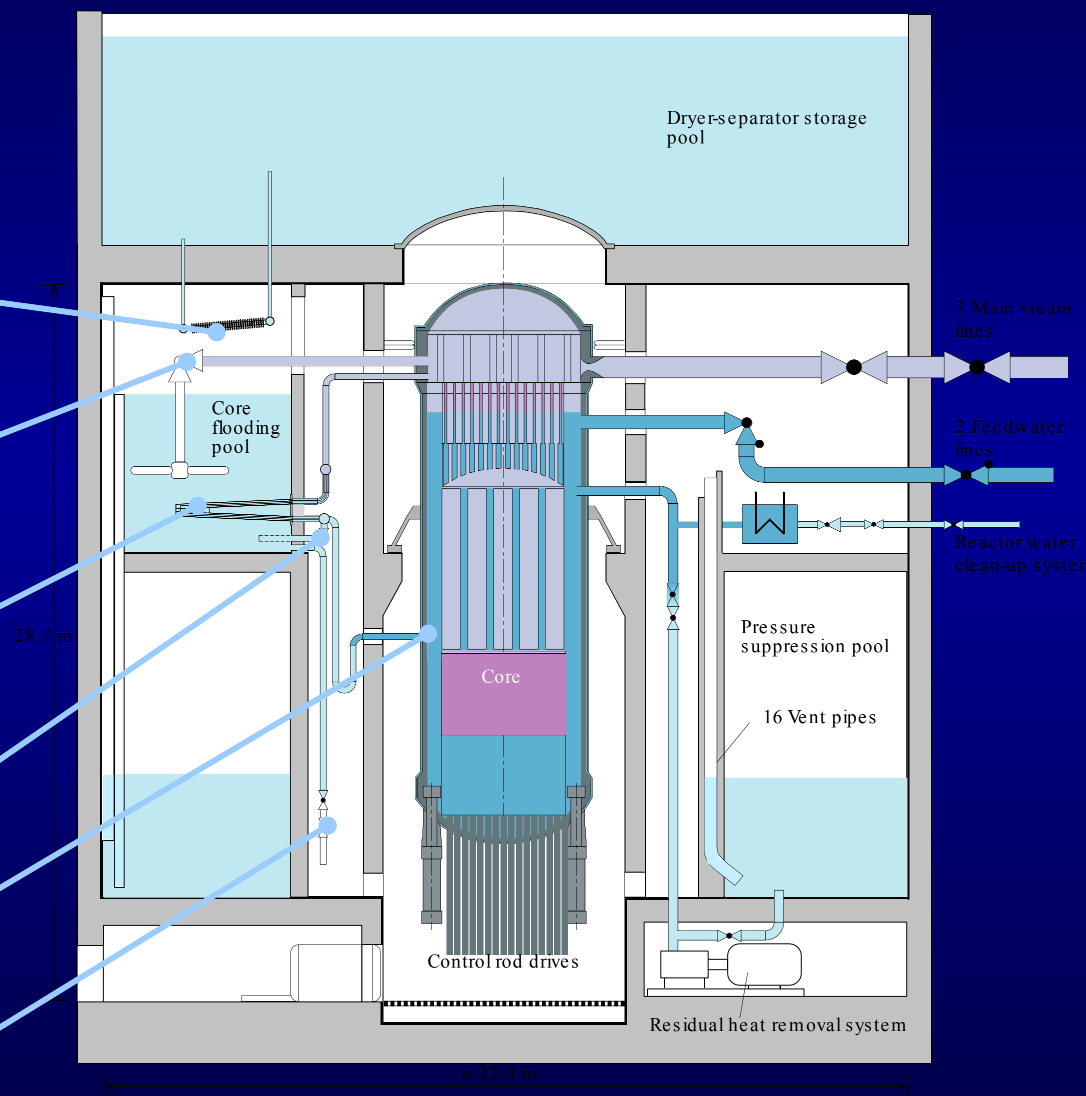
SRV for reactor over-pressure protection and reactor depressurization (6)

Emergency condensers for heat removal from the RPV (4)

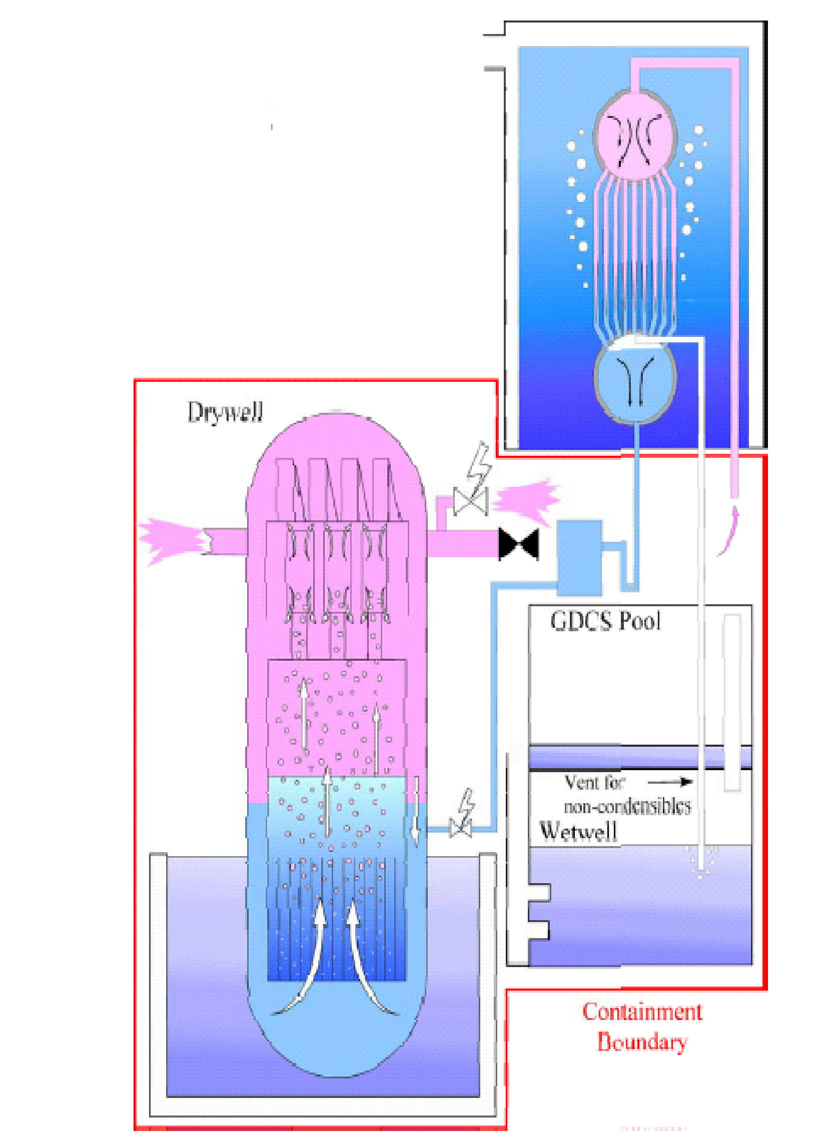
Flooding lines for passive core flooding in the event of LOCA (4)

Passive Outflow Reducer

Drywell flooding line (event of a core melt accident)



ESBWR: Passive Containment Cooling (PCC)



- Initial blowdown energy release transferred to containment heat sink (suppression pool) and PCC heat exchangers.
- Long term decay heat removal is accomplished through PCC heat exchangers. Flows are driven by drywell to wetwell ΔP.
- Condensate drains to holding tank before returning to vessel; non-condensable gas purged to lower wetwell.

ESBWR Thermal-Hydraulic Modeling Issues

- Distribution and effect of non-condensable gas on passive component performance and natural circulation are important processes.
- Passive Containment Cooling (PCC) heat exchangers must condense steam in presence of non-condensable gas. Tests have shown periodic venting of non-condensable gas to wetwell.
- Close coupling between primary and containment. Containment pressure determined by non-condensables in wetwell airspace & vapor pressure.
- Suppression pool condensation, heat transfer & stratification.

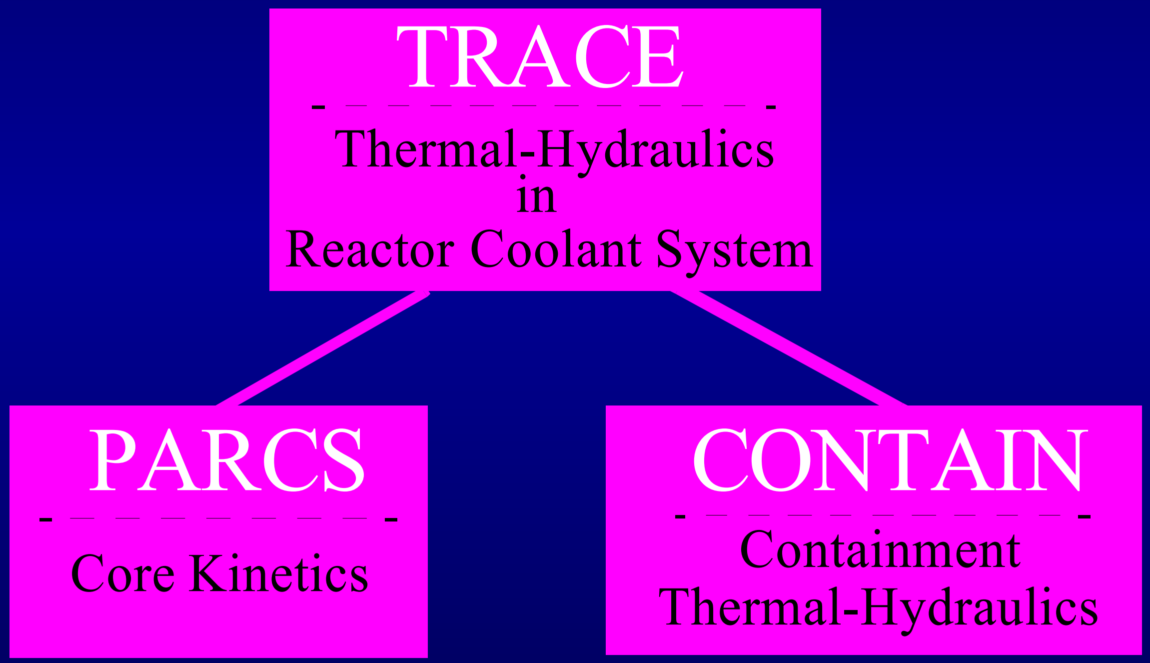
SWR-1000 Thermal-Hydraulic Modeling Issues

- Distribution of non-condensable gas and effect on condensation important. Containment cooling condensers operate with storage pool water circulating through tube bank with external condensation of steam-gas mixture.

- New design features:
  - Passive Pressure Pulse Transmitter (PPPT)
  - Passive outflow reducer to reduce vessel inventory loss
  - Emergency condensers in core flooding pool

NRC Related Development Activities

- NRC thermal-hydraulic code (TRACE) coupled with containment analysis code (CONTAIN) to provide independent evaluation model. This make efficient use of CONTAIN models for suppression pool.



- Model development to improve condensation in presence of non-condensable gas.

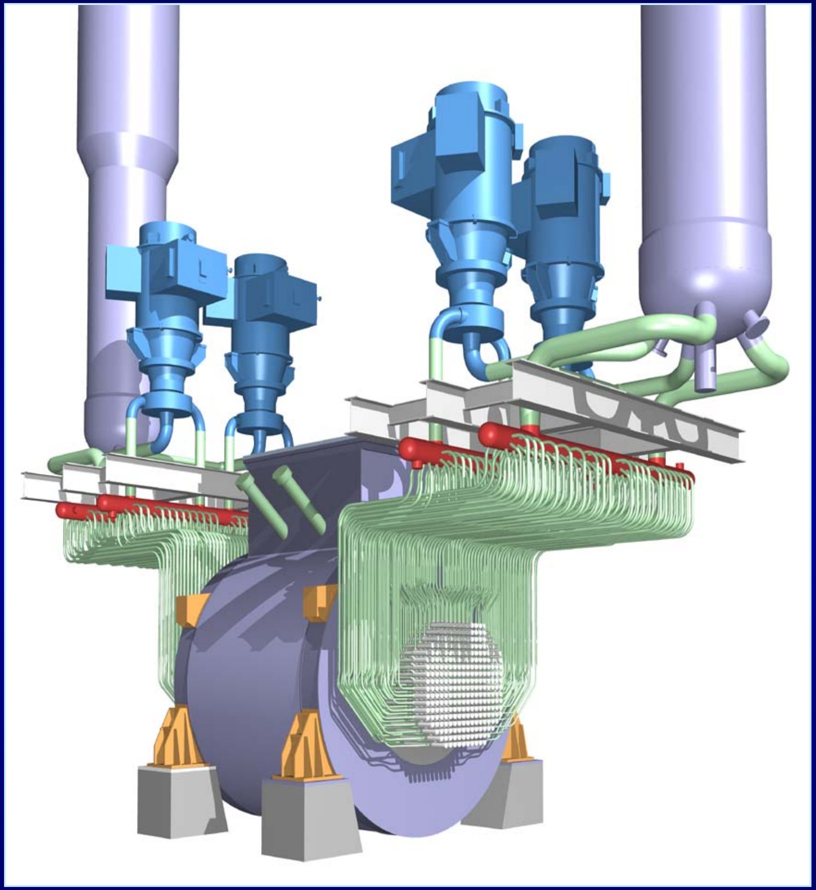
NRC Related Experimental Investigations

- PUMA integral test facility being used to simulate several accident scenarios important to ESBWR; MSLB, GDCS, Bottom drain line break.
- Scaling analysis to identify necessary facility modifications.

ACR - 700

ACR-700: Advanced CANDU Reactor

- 1982 WMt / 731 MWe
- Light-water coolant (in fuel bundles)
- Heavy-water moderator (in calandria)
- Slightly enriched uranium fuel (2%)
- Negative void reactivity coefficient
- Modular horizontal fuel channels
- On-power refueling
- Conventional steam generators (2) and heat transport pumps (4) above core.

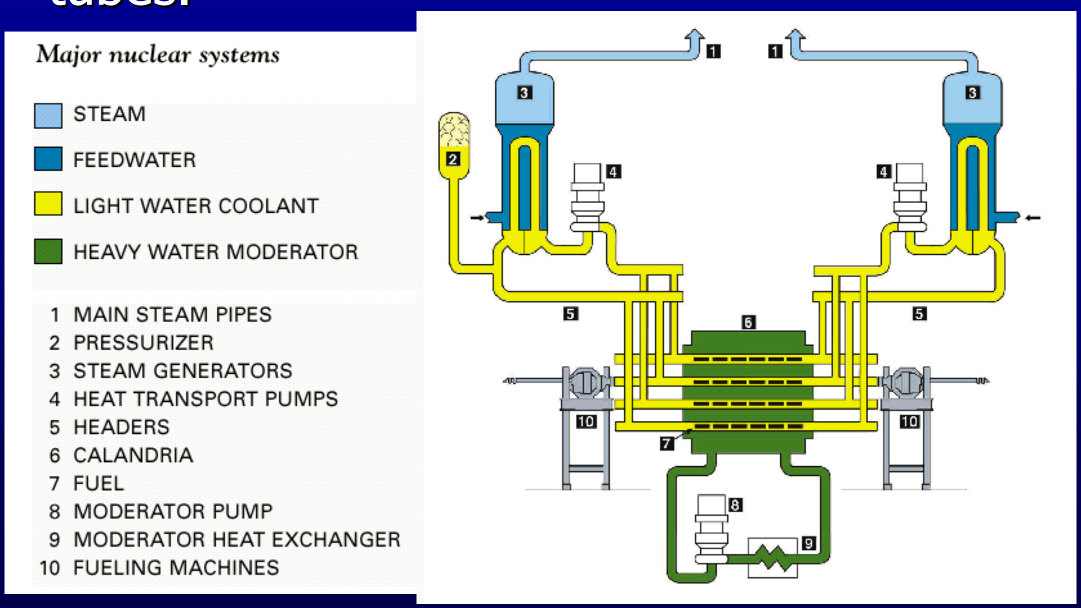


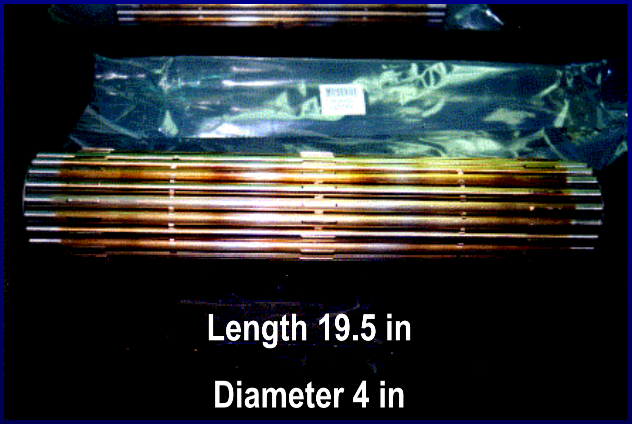
Emergency Core Cooling System

- ❑ **Accumulators for high pressure injection.**
- ❑ **Low pressure pump injection for long term decay heat removal.**

ACR-700 Thermal-Hydraulic Modeling Issues

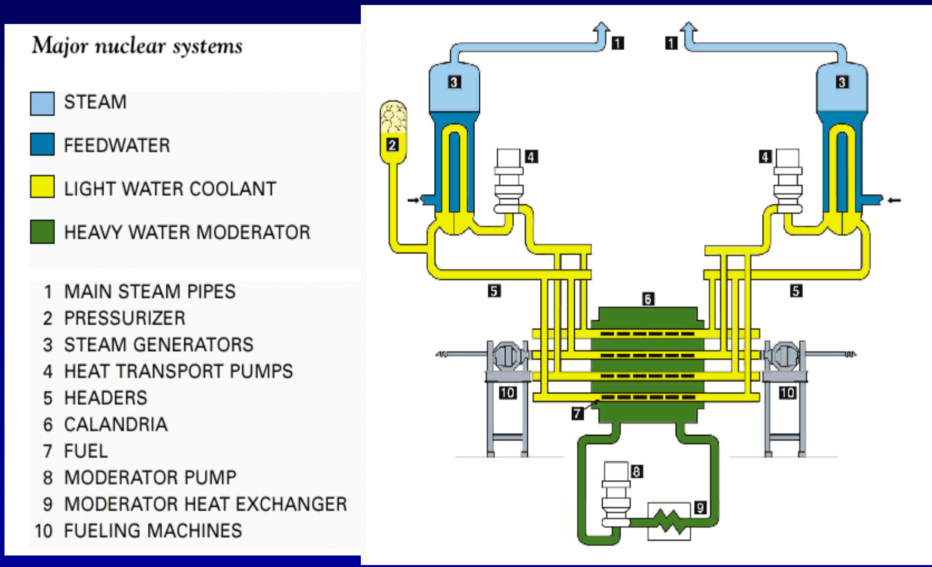
- Existing staff thermal-hydraulic codes (TRACE and RELAP) designed for vertical rod bundles. Need to model/assess codes for horizontal flow patterns and pattern transitions in fuel bundle.
- Lateral quench and rewet processes, formation of dry patches within pressure tube, CHF and post-dryout behavior in fuel.
- Two-phase flow distribution from inlet/outlet headers to pressure tubes.

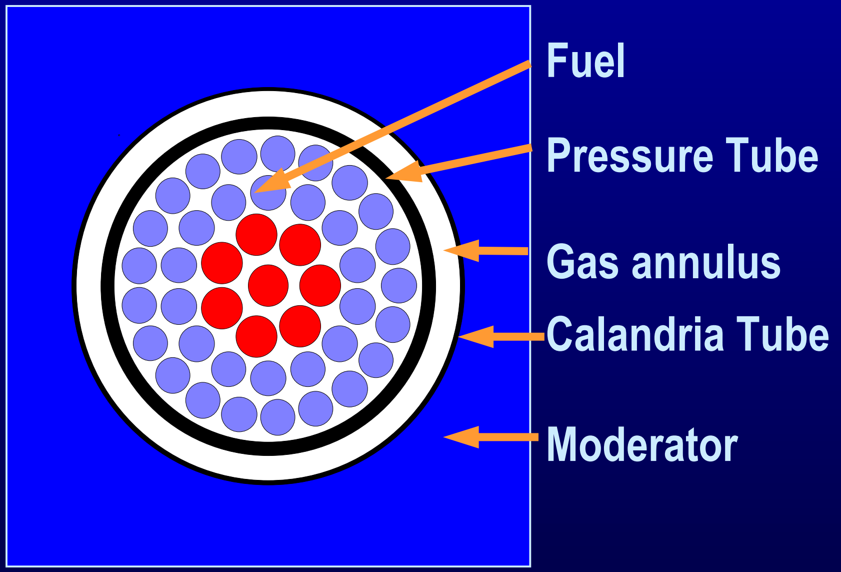




ACR-700 Thermal-Hydraulic Modeling Issues

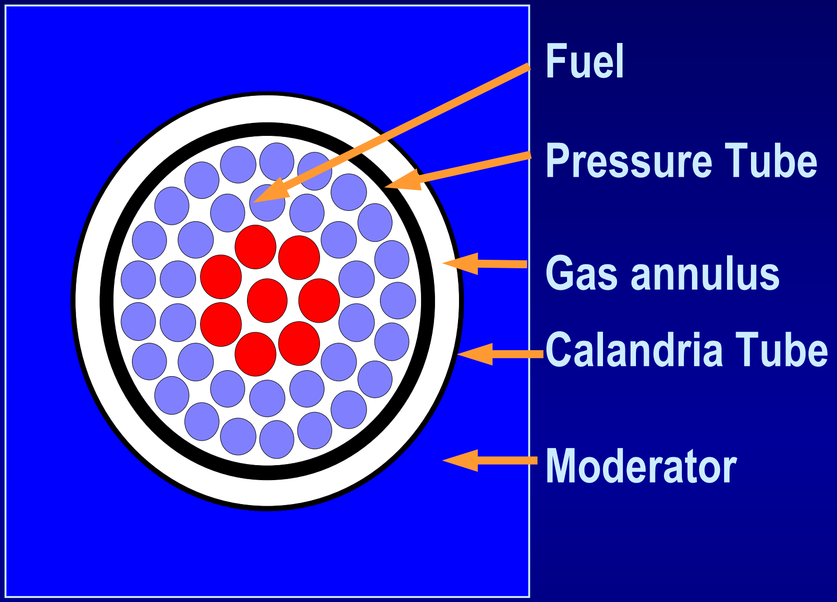
- Existing U.S. thermal-hydraulic codes oriented to vertical rod bundles. Need to model/assess codes for horizontal flow patterns in fuel bundle.
- Modeling moderator thermal-hydraulics, and effect of both light water (coolant) and heavy water (moderator) on reactor kinetics.
- Two-phase flow distribution in inlet/outlet headers.
- Lateral quench and rewet processes, including CHF and post-dryout behavior in fuel.





ACR-700 Thermal-Hydraulic Modeling Issues

- Heat transfer between pressure tube and calandria tube. Pressure tube “sagging” may occur during some accident scenarios, resulting in contact between pressure tube and calandria tube.



- Potential need for severe accident data for:
  - **Melt relocation stages in horizontal core geometry**
  - **Heat transfer to moderator from sagging fuel and pressure tubes**

NRC Related Activities

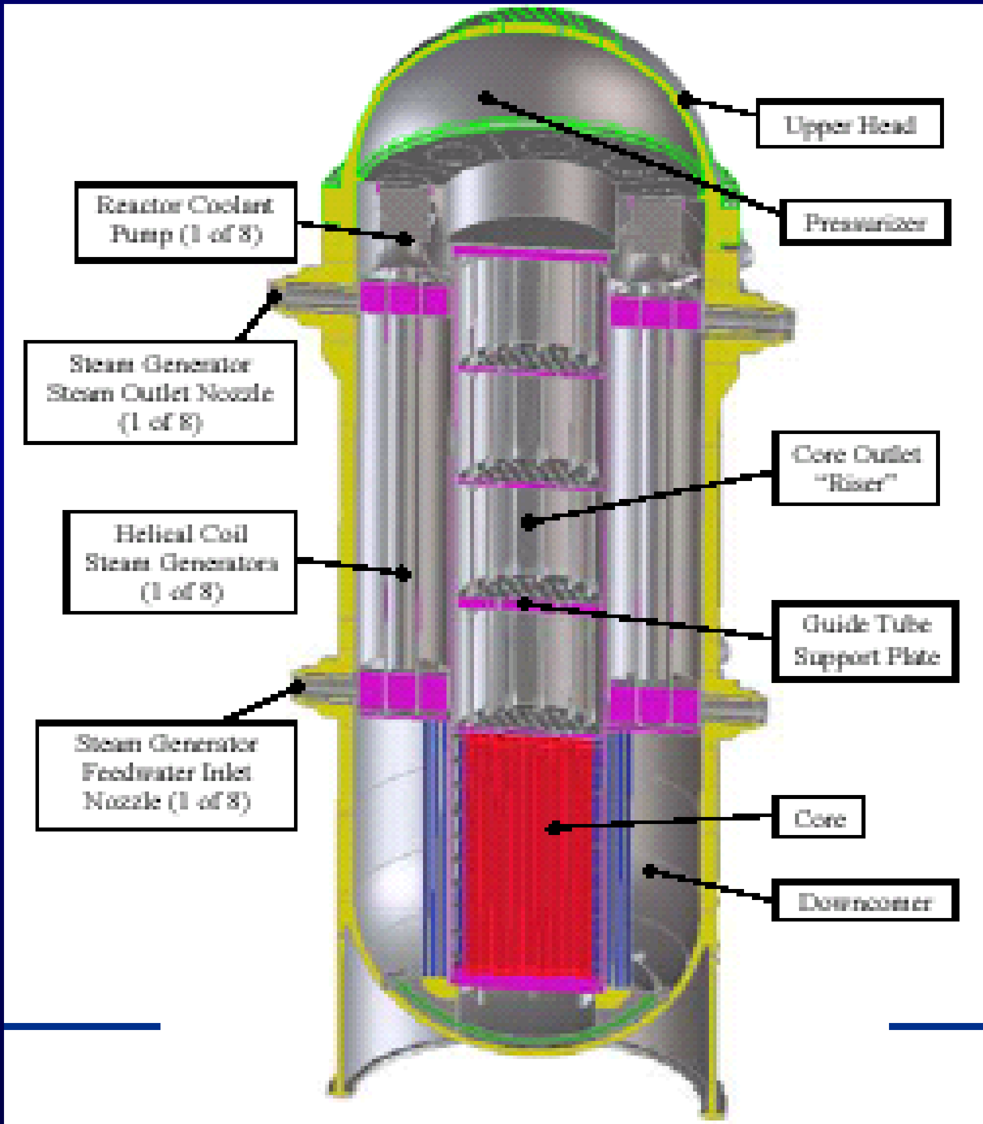
- NRC staff is working with AECL to make effective use of existing database for CANDU reactors. I
- Independent PIRT (Phenomena Identification and Ranking Table) being developed to guide code development & define test needs.
- Scaling analysis for RD-14, RD-14M and RD-14/ACR integral test facilities.
- PIRT and scaling evaluation to determine need for new experimental tests and/or facilities.

IRIS

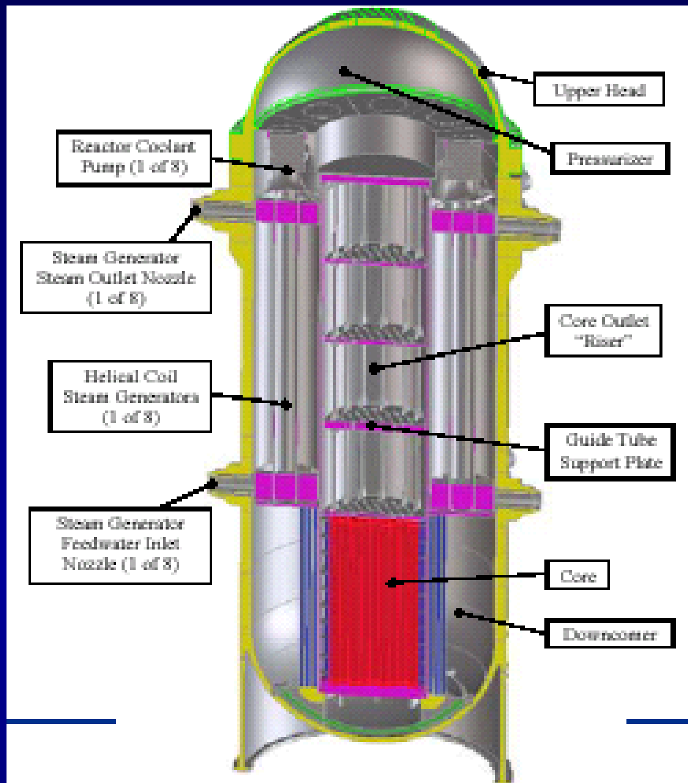
International Reactor Innovative and Secure

IRIS

- Integral LWR (300 MWe)
- Safety by innovative design and passive safety systems
- Multiyear straight burn core ?
- Integral helical coil, once through SGs with superheat.
- Spherical Containment
- Meets Generation IV Objectives
  - Proliferation Resistance
  - Enhanced Safety
  - Improved Economics
  - Reduced Waste

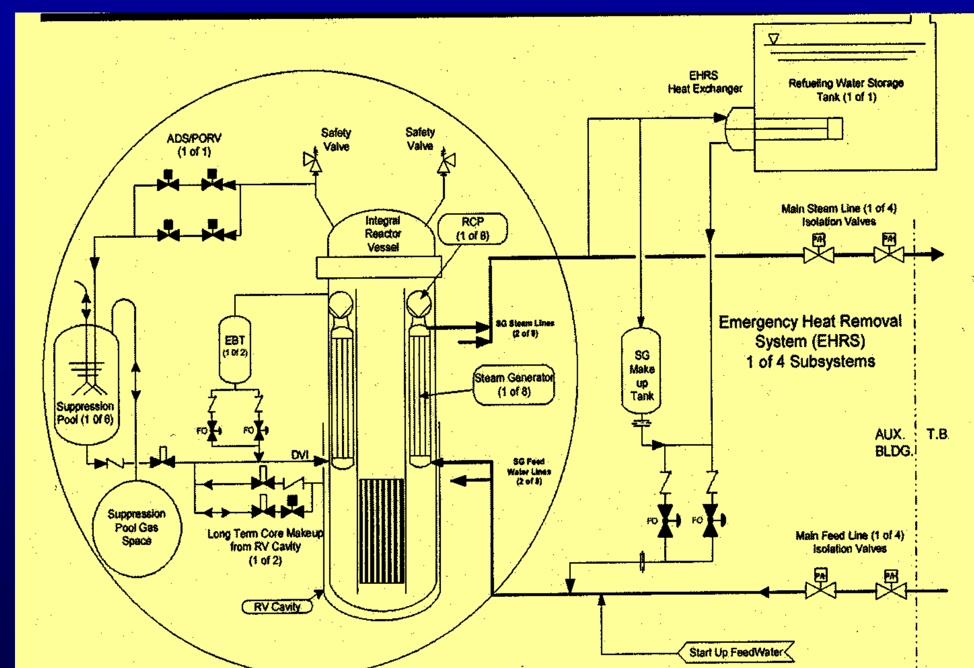


IRIS Thermal-Hydraulic Modeling Issues



- Helical coil once-through steam generators are unique. Two-phase flow patterns and transitions need to be understood and modeled.

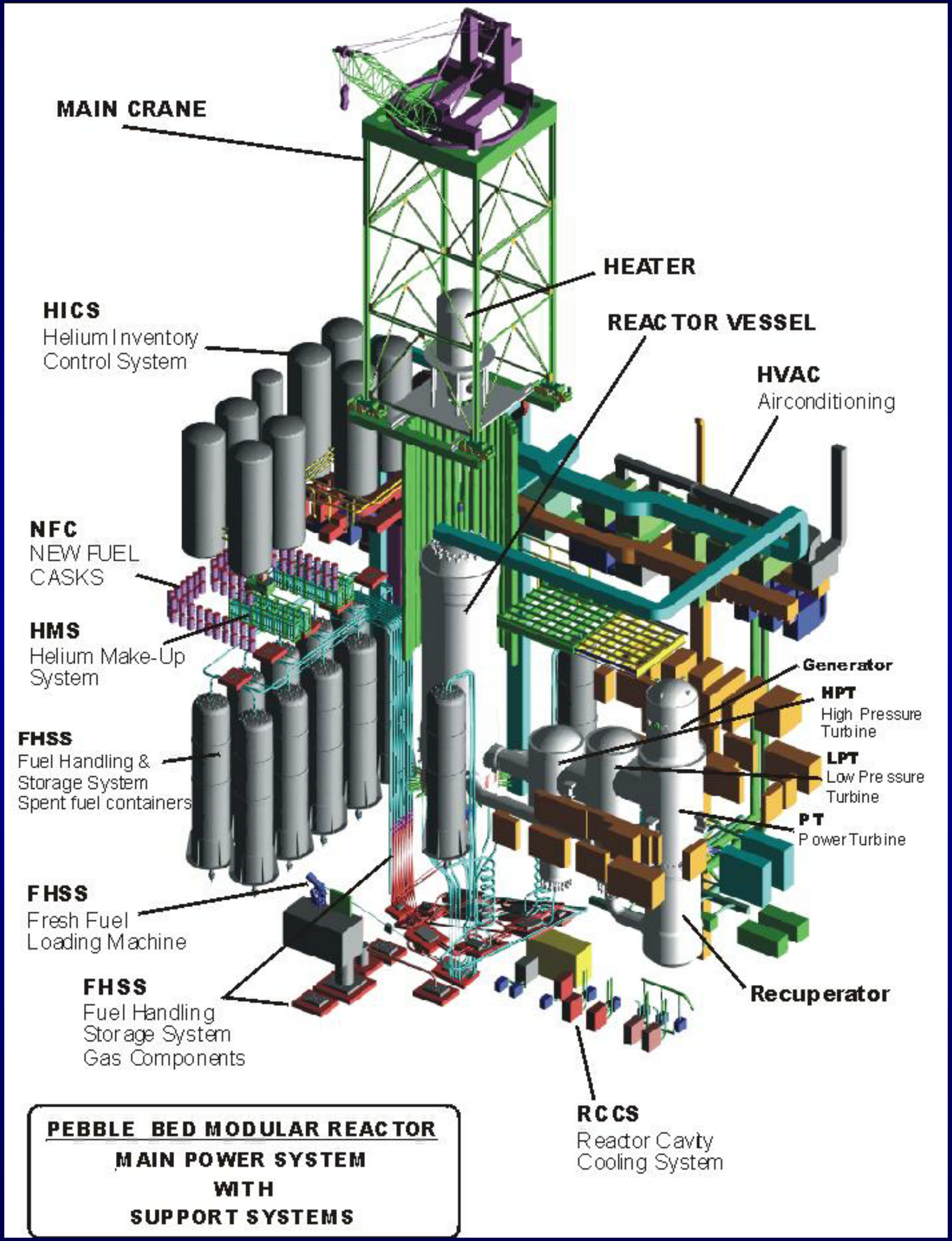
- Primary and containment represent a tightly coupled system. IRIS containment is designed for high pressure (12 bar / 175 psig), includes suppression pool as a source of gravity driven, borated water.



Gas Reactors

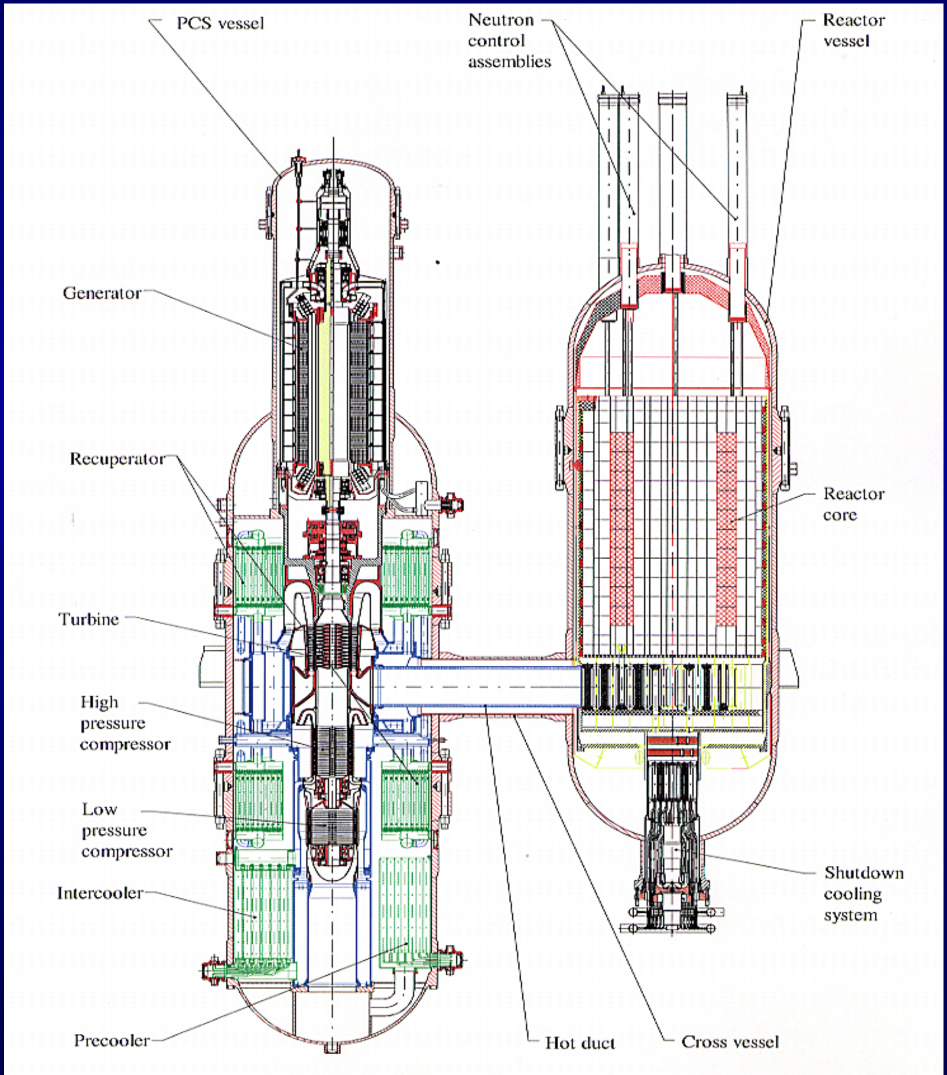
Pebble Bed Modular Reactor (PBMR)

- **High Temperature Helium Cooled Reactor**
- **165 MWe range per module**
- **8 modules per common control room**
- **Coated Particle Fuel**
- **Spherical Fuel Elements (as per German reactors)**
- **10 years fuel storage in plant**
- **Direct Cycle Gas Turbine (multi-shaft)**
- **No "safety systems"**
  - **fuel integrity maintained under most severe possible accident**
- **Exelon decided not to continue with preapplication**
  - **PBMR Pty. planning preapplication review**



Gas-Turbine Modular Helium Reactor (GT-MHR)

- International Design (General Atomics, etc.) for US/Russian Pu disposition strategy
- Pre-application review underway
- 286 MWe per module
- 4 modules per common control room
- Hexagonal prismatic blocks similar to Fort St. Vrain Design
- TRISO ceramic particle fuel



NRC Approach and Related Activities

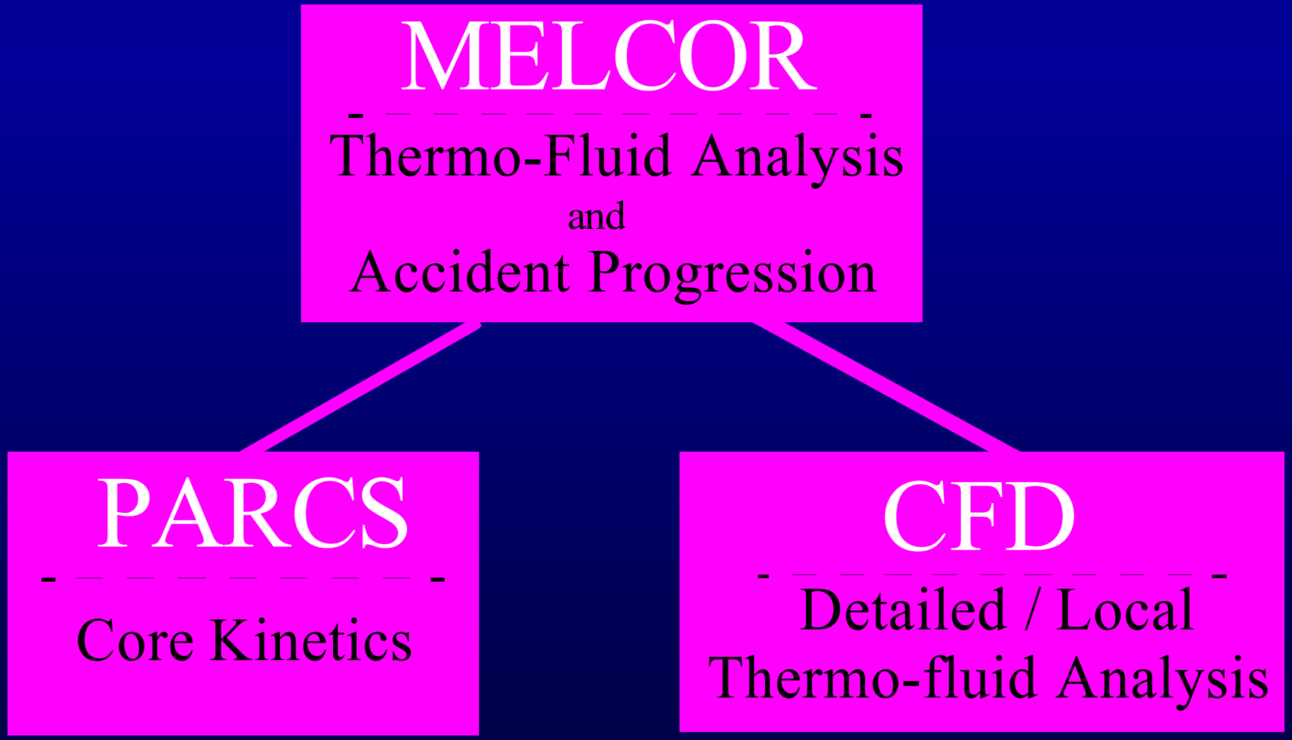
- New regulatory framework is necessary for gas reactors, and approach is likely to be "risk-informed."

- PIRT process to be applied to identify code development and experimental testing requirements.

  - **Modeling heat transfer & fluid flow in packed bed**
  - **Conduction & radiation modeling / spherical geometry for PBMR fuel**
  - **High temperature properties of graphite**
  - **Tracking of multiple non-condensable gas species (He and air)**
  - **HTGR Turbo-machinery components**

NRC Analysis Approach

- GRSAC code obtained & being used for initial studies.
- Long-range plan is to couple severe accident code (MELCOR) and kinetics code (PARCS) for HTGR analysis. CFD code to be used/coupled for detailed calculations.



Summary

- There has been renewed activity and interest in advanced reactors in last few years. Several new designs are under consideration.

- Evaluation of these new applications (ESBWR, SWR-1000, ACR-700, IRIS, PBMR, GT-MHR) will likely require experimental work and code development to handle unique thermal-hydraulic issues.

Nuclear Safety Research Conference, 2003

"Regulatory Process and Structure for Future Nuclear Power Plant Licensing"

Mary Drouin, Tom King
U.S. Nuclear Regulatory Commission

John Lehner, Trevor Pratt, Vinod Mubayi
Brookhaven National Laboratory

October 21, 2003

OUTLINE

- Why a new regulatory structure?
- What is this new regulatory structure?

WHY A NEW REGULATORY STRUCTURE?

- Why is there a need for a new regulatory structure for advanced reactors?
- What should be the framework of this regulatory structure?

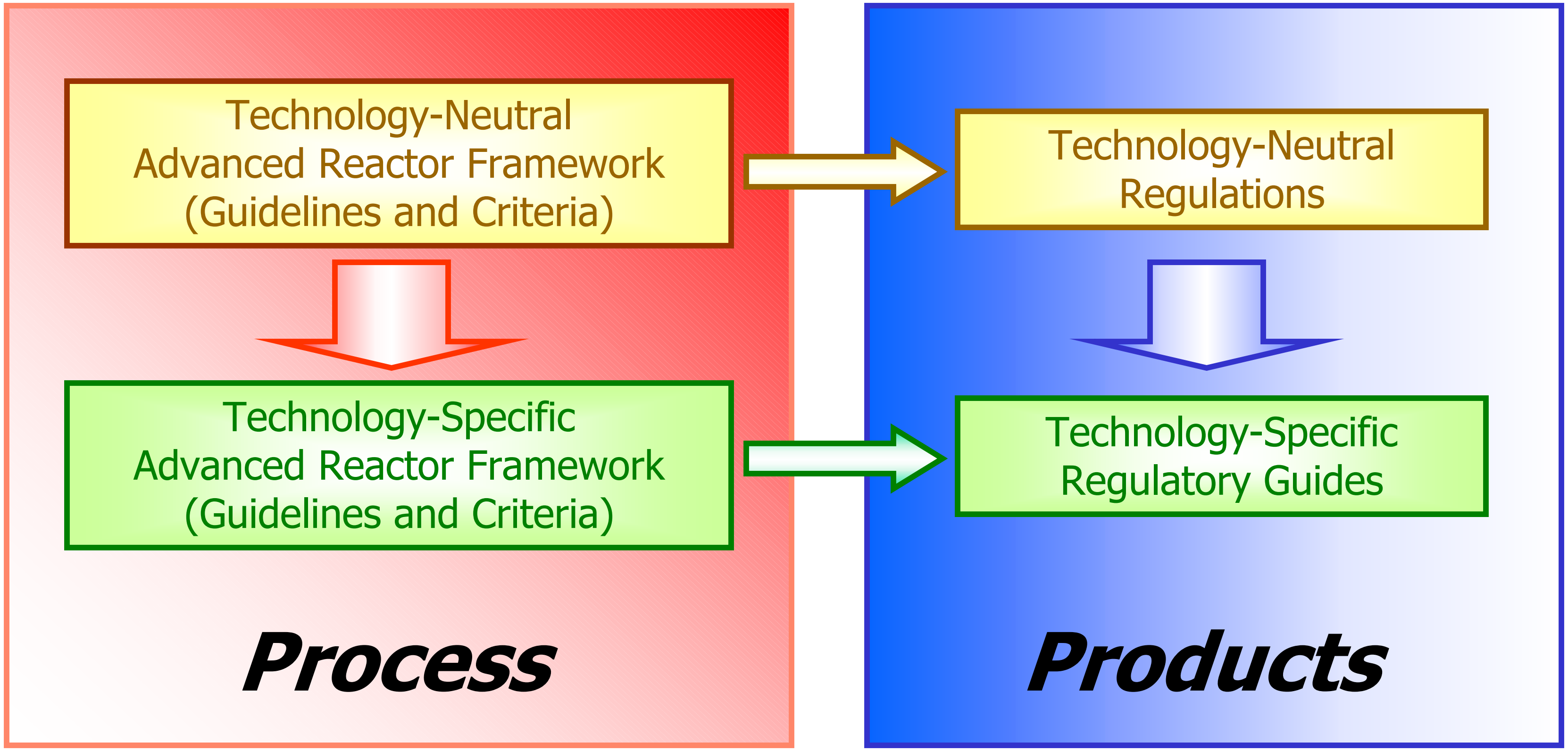
WHY A NEW REGULATORY STRUCTURE? (cont'd)

- Current set of regulations in 10 CFR Part 50 are based on light water reactor (LWR) technology
- There will be design and operational issues associated with the advanced reactors that are distinctly different from current LWRs issues
- Current LWRs has evolved over five decades, and the bulk of this evolution occurred without the benefit of insights from probabilistic risk assessments (PRAs)
- Using current regulatory process less efficient and effective

WHY A NEW REGULATORY STRUCTURE? (cont’d)

- Ensure that a structured and systematic approach is used during the development of the regulations that will govern the design and operation of advanced reactors
- Ensure a more systematic selection of performance measures to use in regulation
- Ensure uniformity, consistency, and defensibility in the development of the regulations, particularly when addressing the unique design and operational aspects of advanced reactors

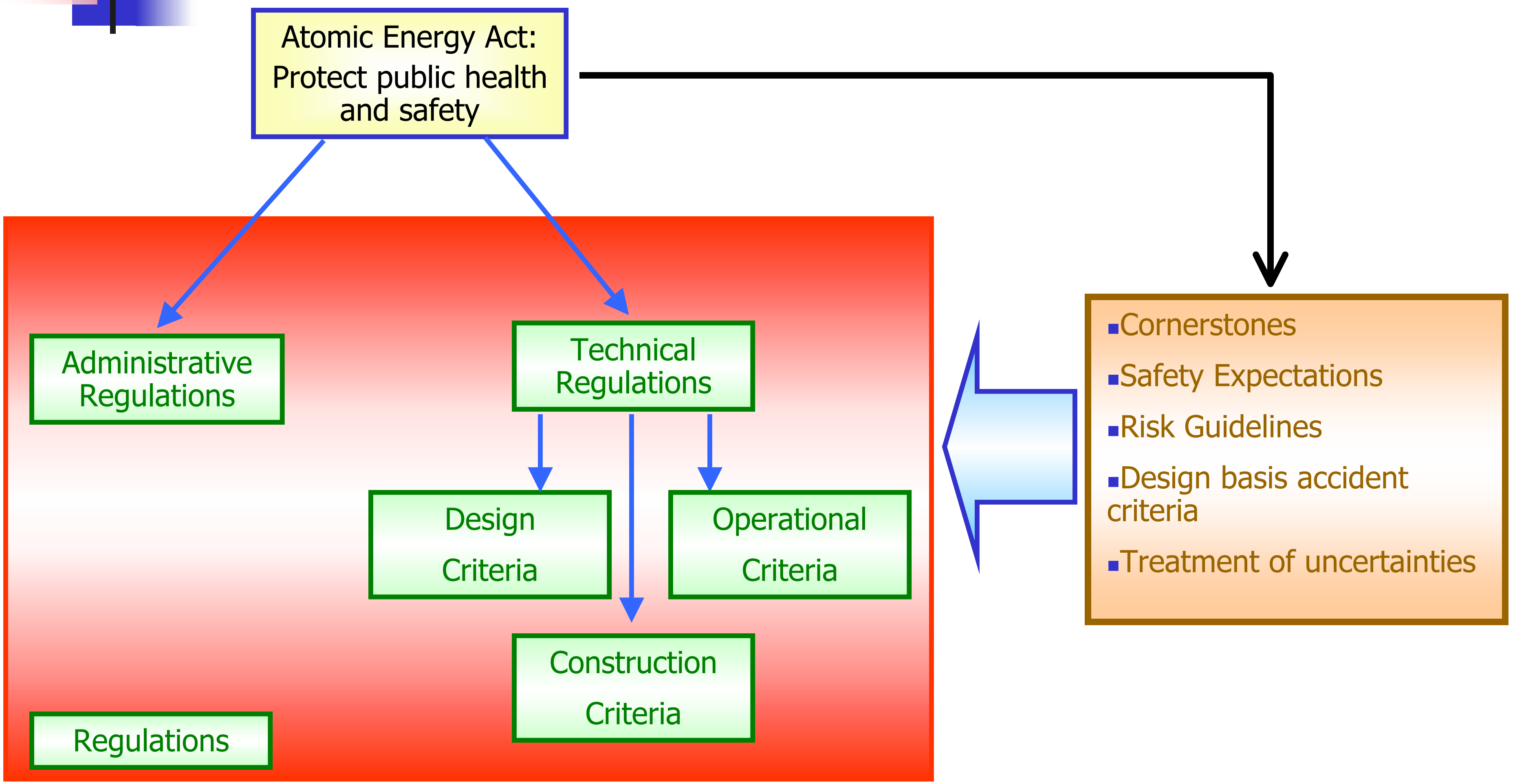
NEW REGULATORY STRUCTURE



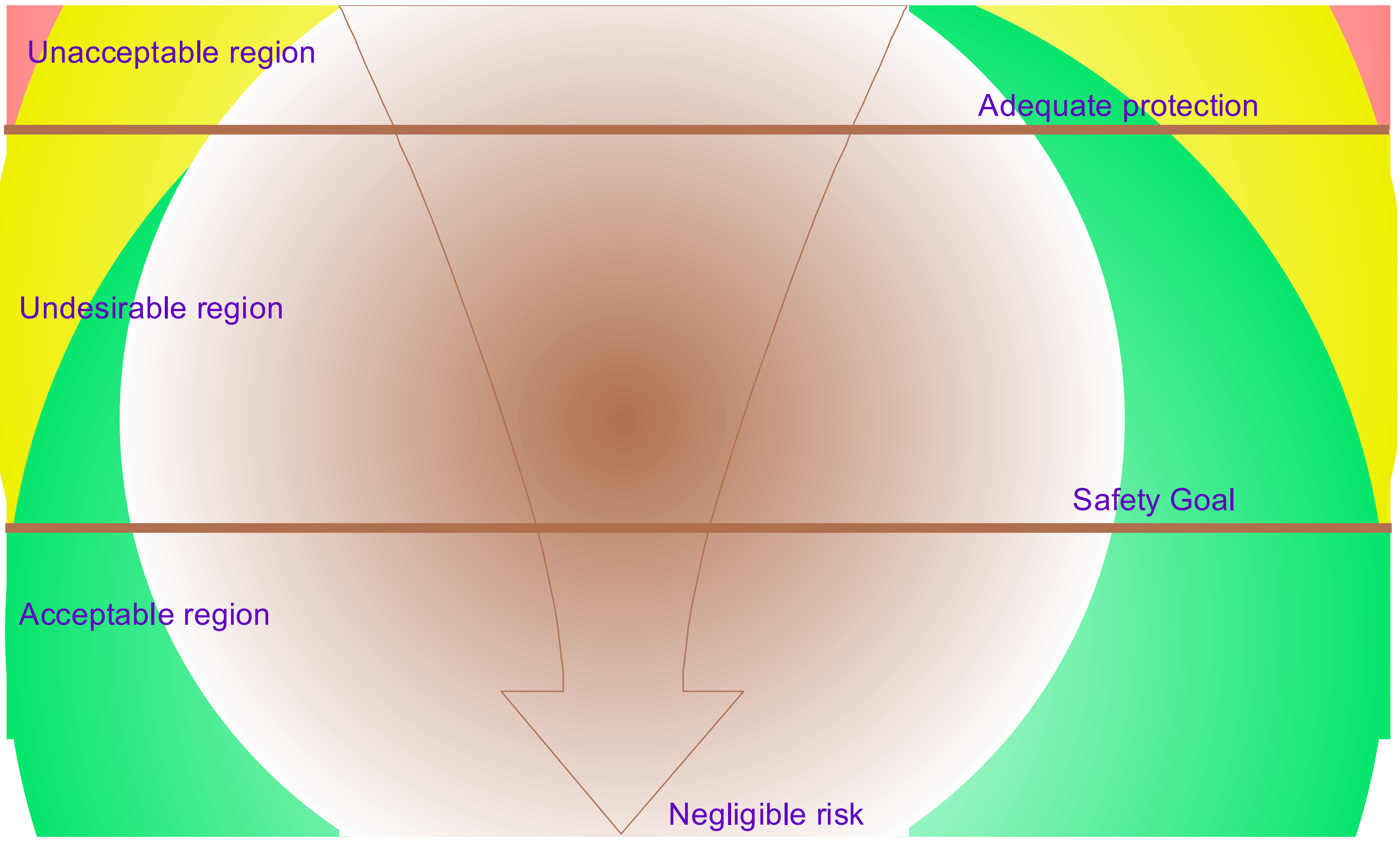
TECHNOLOGY-NEUTRAL FRAMEWORK TECHNICAL ISSUES

- Establish the ***process*** for deriving the technology-neutral regulations
- Establish guidance for the ***safety expectations*** of future reactors
- Establish the ***risk guidelines*** commensurate with the safety expectations
- Identify the necessary ***cornerstones*** for safe nuclear power plant design, construction and operation for future reactors
- Develop ***design basis accident criteria*** for the design, construction and operation of advanced reactors
- Develop the guidance for ***treatment of uncertainties*** in terms of defense-in-depth concept

TECHNOLOGY NEUTRAL FRAMEWORK: PROCESS (preliminary)

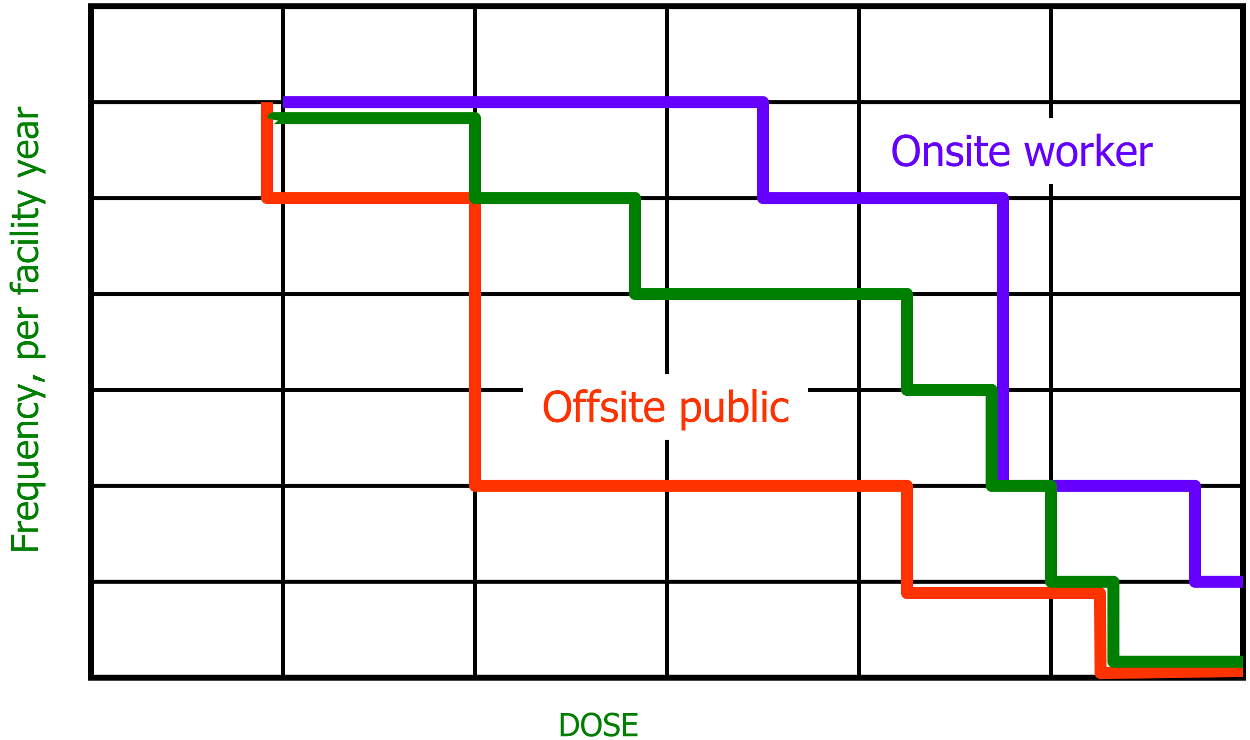


TECHNOLOGY NEUTRAL FRAMEWORK: SAFETY EXPECTATIONS (preliminary)

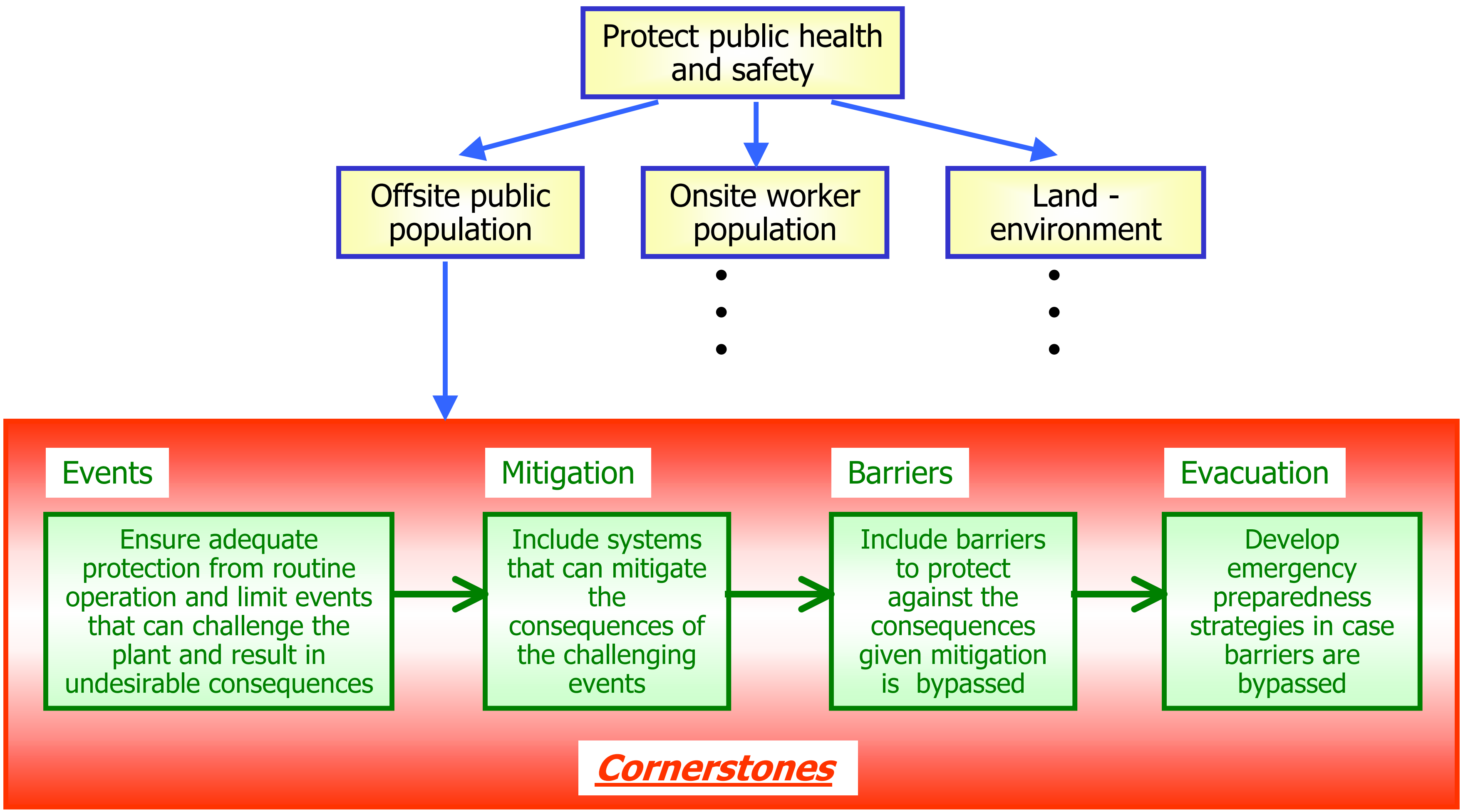


TECHNOLOGY NEUTRAL FRAMEWORK: RISK GUIDELINES (preliminary)

- Safety goal for advanced reactors: tie risk criteria to current quantitative health objectives (Early: 5*10-7, Latent: 2*10-6)



TECHNOLOGY NEUTRAL FRAMEWORK: CORNERSTONES (preliminary)



TECHNOLOGY NEUTRAL FRAMEWORK: DESIGN BASIS ACCIDENT CRITERIA (preliminary)

- DBAs identified for each event group (i.e., frequent, infrequent and rare events)
  - Most risk significant event, based on plant-specific PRA results, for each group identified as the DBA
- Acceptance criteria for DBA
  - Risk guidelines
  - Deterministic dose guidelines
- Design must comply with defense-in-depth criteria; e.g.,
  - Not rely on single cornerstone
  - Not rely on single function/system within each cornerstone

TECHNOLOGY NEUTRAL FRAMEWORK: TREATMENT OF UNCERTAINTIES (preliminary)

- Uncertainties are treated through defense-in-depth:
  - Identify the different sources and types of uncertainties
  - Establish the different elements of defense-in-depth
  - Establish both probabilistic and deterministic (i.e., rationalist and structuralist) criteria
  - Develop model that matches the different defense-in-depth element to treat each source of uncertainty based on both the probabilistic and deterministic criteria

NEXT STEPS . . . .

- Schedule for initial/preliminary draft:
  - technology neutral framework 12/03
  - technology neutral regulations 6/04
  - technology specific framework 2005
  - technology specific regulatory guides 2006
- Stakeholder Interactions
  - Continue to hold public meetings and workshops
  - Continue to interface with international organizations

Generation IV Concepts and Technology Challenges

NRC

Nuclear Safety Research Conference

October 21, 2003

Jason Remer
Program Director, Next Generation Systems Development
Advanced Nuclear Research
Office of Nuclear Energy, Science and Technology

Overview

- Nuclear Power Timeline
- Generation IV Initiative
- DOE Priorities
- Next Generation Nuclear Plant
- Supercritical Water Cooled Reactor
- Fast Reactors for Sustainability
- Long Term Strategy
- Summary

The Development of Nuclear Power -- Past, Present and Future

Generation I

Early Prototype Reactors

- Shippingport
- Dresden, Fermi I
- Magnox

Generation II

Commercial Power Reactors

- LWR-PWR, BWR
- CANDU
- VVER/RBMK

Generation III

Advanced LWRs

- ABWR
- System 80+
- AP600
- EPR

Near-Term Deployment

Generation I-III Evolutionary Designs Offering Improved Economics

Generation IV

- Highly Economical
- Enhanced Safety
- Minimal Waste
- Proliferation Resistant

Gen I | Gen II | Gen III | Gen III+ | Gen-IV

1950 | 1995 | 2000 | 2005 | 2010 | 2015 | 2020

Generation IV Initiative

- **U.S. Generation IV Initiative established in 2000 as an international effort**
- **Program designed to lead development of advanced reactors to a state of maturity allowing for commercial deployment by 2015 or later**
- **Generation IV reactors will offer improvements in**
  - Reactor safety and reliability
  - Proliferation resistance and physical protection
  - Economics compared to existing reactors
  - Sustainability
- **Six reactor concepts were selected for further research in the Generation IV Technology Roadmap (issued Dec 2002)**
  - Gas-cooled Fast Reactor (GFR)
  - Lead-cooled Fast Reactor (LFR)
  - Molten Salt Reactor (MSR)
  - Sodium-cooled Fast Reactor (SFR)
  - Supercritical Water-cooled Reactor (SCWR)
  - Very High Temperature Reactor (VHTR) -- Next Generation Nuclear Plant (NGNP)

U.S. DOE Generation IV Priorities

- **Gen IV “A”**
  - VHTR
  - SCWR

→ NGNP

- **Gen IV “B”**
  - GFR
  - LFR
  - SFR

→ **U.S. Fast Reactor ?**

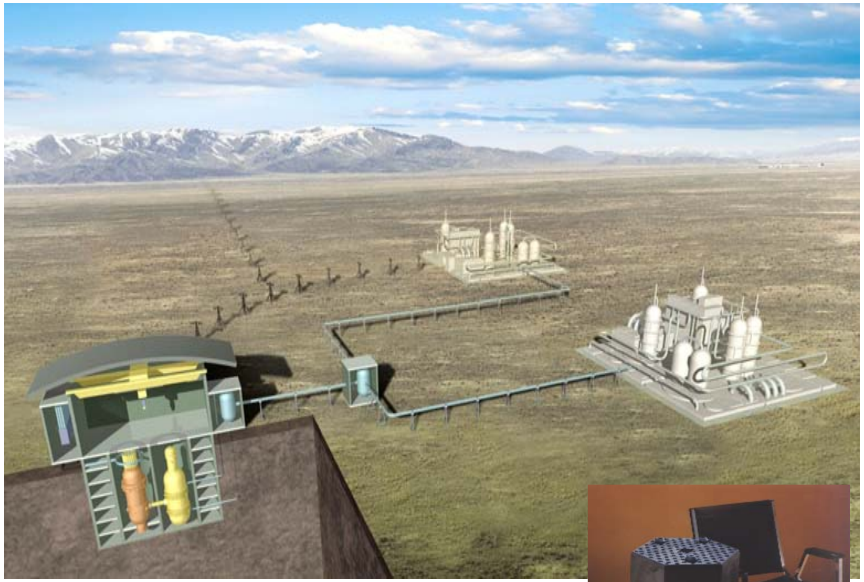
- **Closely coordinated with Advanced Fuel Cycle Initiative**

Requirements for A Next-Generation Nuclear Plant (NGNP) Project

- Collaborative with international community
- Collaborative with industry, especially utilities
- Demonstrate H_2 and direct-cycle electricity production
- Result in a commercially viable plant design

Next Generation Nuclear Plant (NGNP)

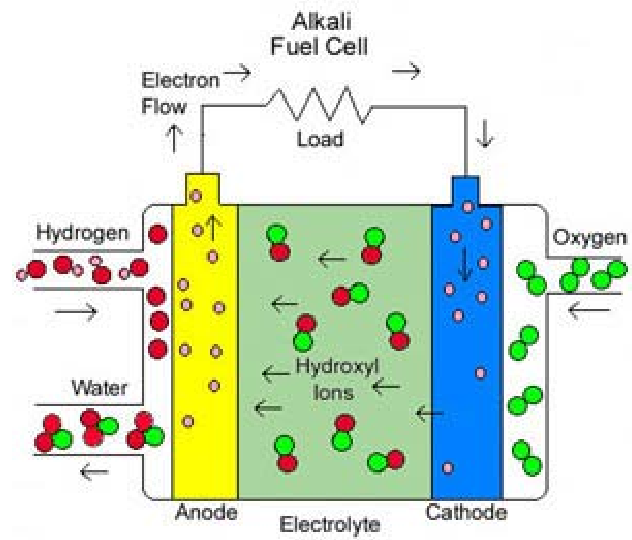
- **NGNP, using Very-High Temperature Reactor (VHTR), is optimized for the co-generation of hydrogen and electricity**
  - High outlet temperature (900-1000°C) allows use of thermochemical and temperature-assisted electrolysis methods for producing hydrogen
  - High electrical conversion efficiency
  - Attractive safety aspects
- **Modular construction**
  - 600 MWTh
  - Solid block graphite core
  - At 50% efficiency, could produce up to 200 MT of H2 a day, the equivalent of 200,000 gallons gasoline per day.

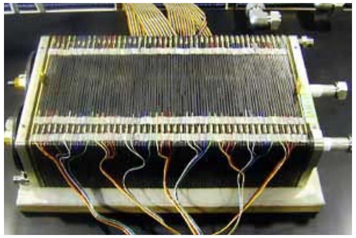


- **Objective: build NGNP demo plant by the middle of the next decade**

NGNP and the Hydrogen Economy

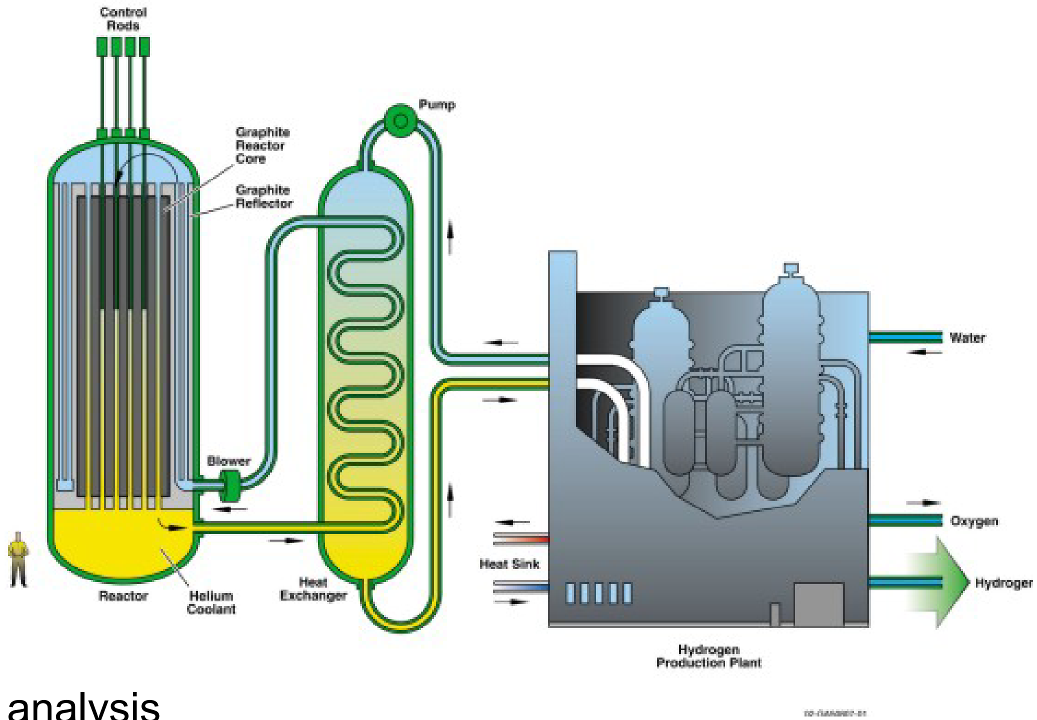
- **Hydrogen is future fuel for transportation sector**
  - Produces no noxious emissions when burned or consumed in fuel cells
  - Can be produced using a number of energy sources (fossil, hydro, solar, nuclear, etc.)
- **Nuclear power can produce hydrogen emission-free to meet the needs of the transportation sector**
- **Supports National Energy Policy, FreedomCAR Initiative, and Nuclear Hydrogen Initiative**





Very-High Temperature Reactor System (VHTR)

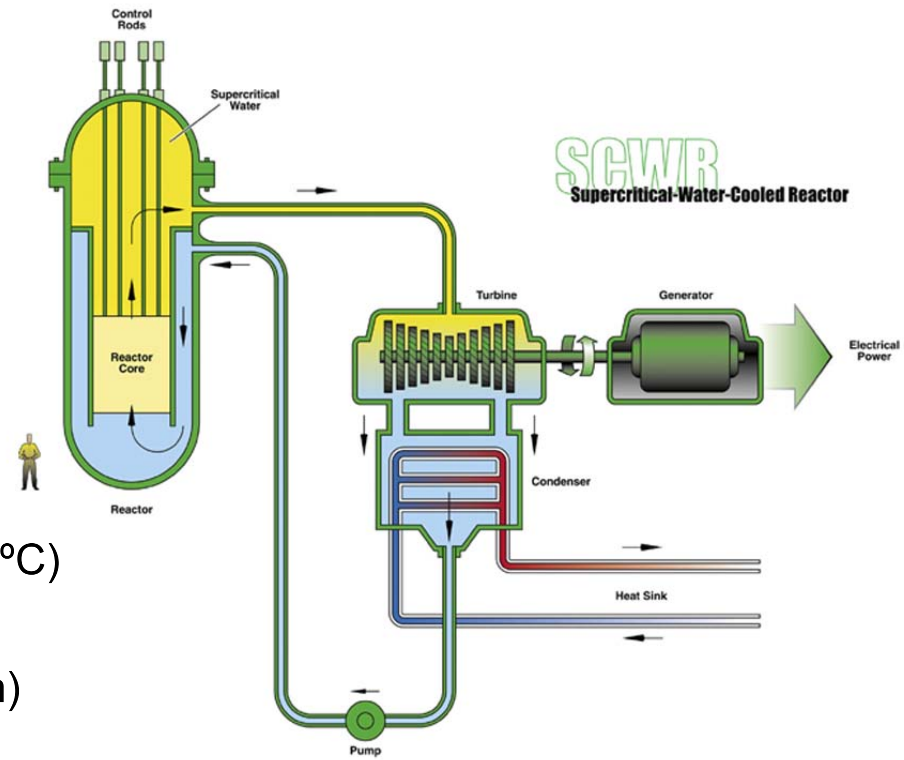
- **VHTR Technology Gaps**
  - Increase core outlet temperature from 850°C to 1000°C
  - Maximum fuel temperature of 1800°C during accidents
  - Advanced materials
  - Avoid power peaking and temperature gradients in the core
- **VHTR Safety R&D**
  - Passive heat removal systems
  - Analysis and demonstration of inherent safety features
  - Design basis and severe accident analysis
  - Fuel development - TRISO-coated particles



Supercritical Water Cooled Reactor (SCWR)

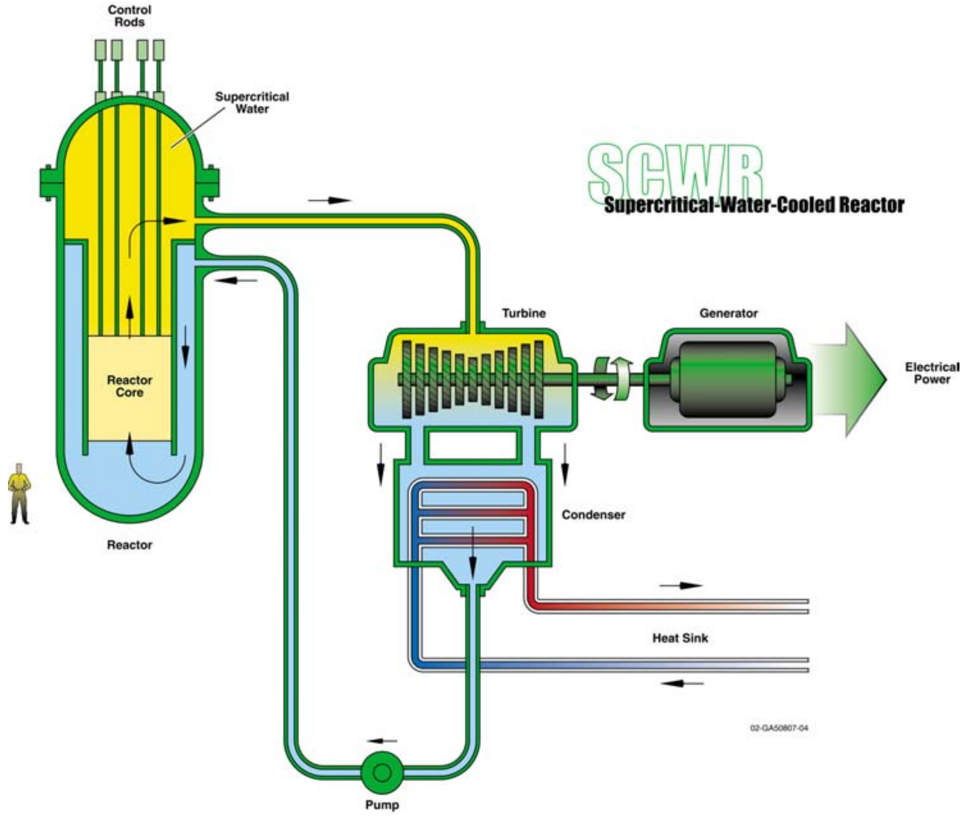
♦ **SCWR: An advanced water reactor for competitive electricity production in the long term**

- Operates above critical point of water (374°C, 22.1 MPa),
- A direct-cycle reactor….without the boiling
  - no steam generator
  - no steam dryer
  - no recirculation pumps
- Economical
  - small containment
  - few major components
  - high efficiency (outlet temperature 510 ºC)
- Flexibility of application
  - electricity production (thermal spectrum)
  - actinide management (fast spectrum)
- Fueled by conventional LEU fuel



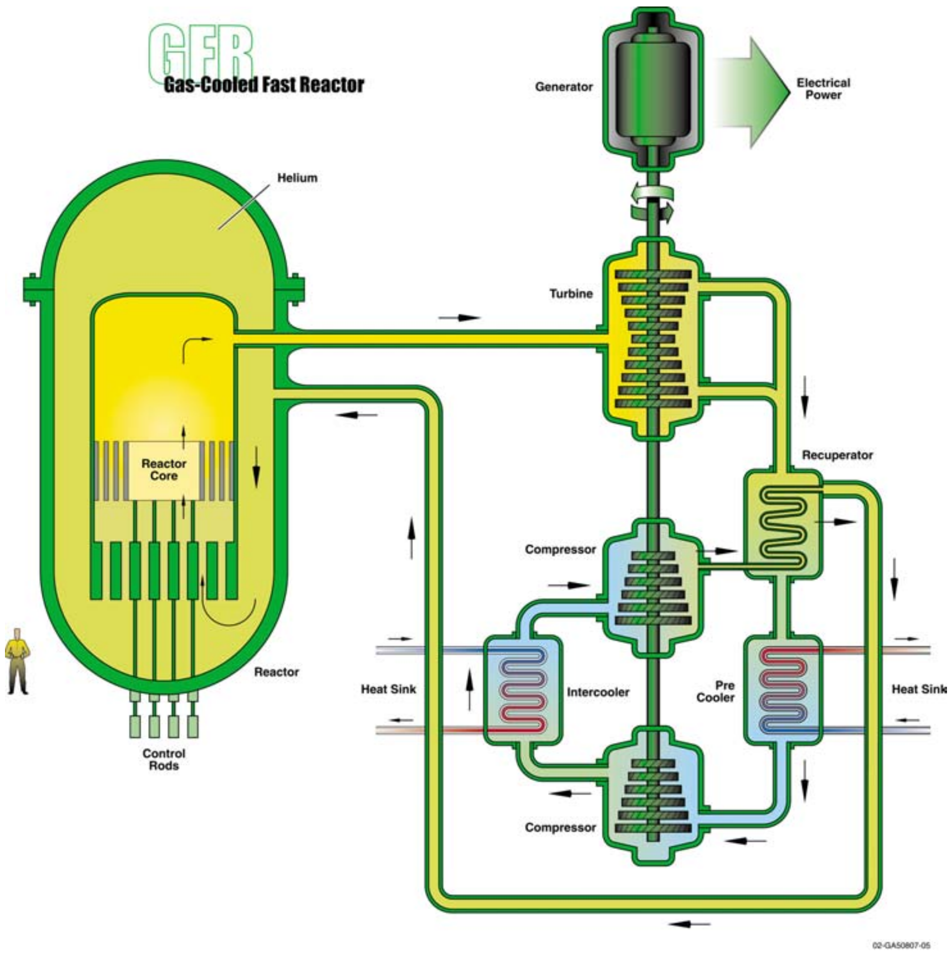
Supercritical Water Cooled Reactor (SCWR)

- **SCWR Technology Gaps**
  - Materials and structures - corrosion and stress corrosion cracking
  - Radiolysis and water chemistry
  - Dimensional and microstructure stability
  - Strength, embrittlement, and creep resistance
  - System pressure / component size at 25 MPa (3600 psi)
- **SCWR Safety R&D**
  - Safety system design - AFW/EFW
  - Power - flow stability



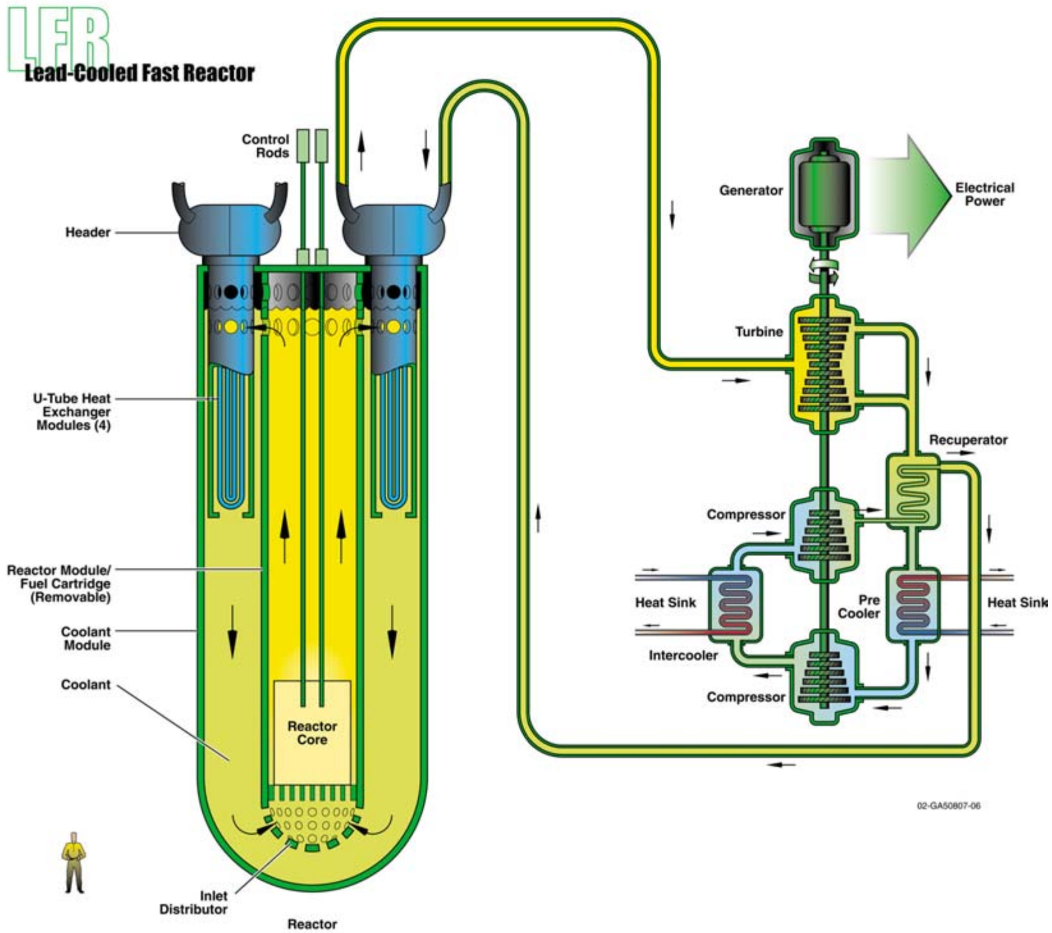
Gas-Cooled Fast Reactor (GFR)

- **GFR Characteristics**
  - Helium coolant (850 ºC outlet)
  - Direct helium turbine conversion cycle, projected 48% efficiency
  - 600 MW_{th}/288 MW_e
  - Several fuel options and core configurations
- **GFR Uses/Benefits**
  - Electricity production, actinide management, possible hydrogen production
- **GFR R&D Challenges**
  - Fuel for fast spectrum
  - Decay heat removal (~100MW_{th}/m^3)
  - Power conversion system
  - Materials for fast neutron fluence under high temperatures



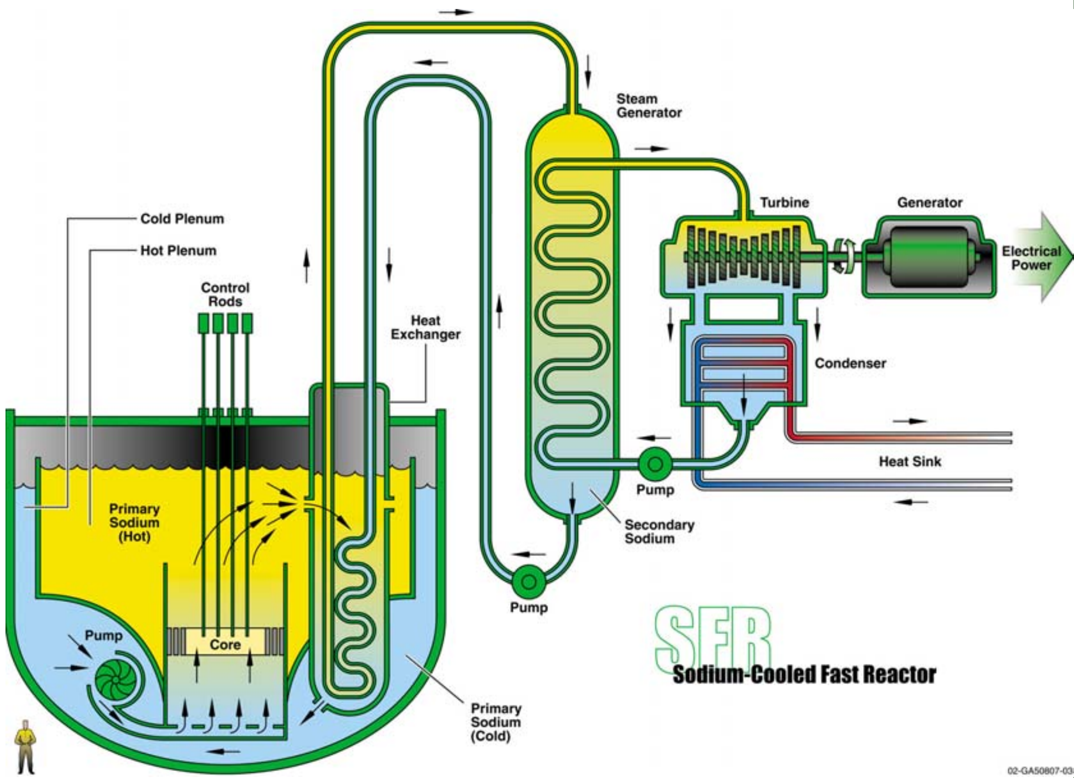
Lead-Cooled Fast Reactor (LFR)

- **LFR Characteristics**
  - Pb or Pb/Bi coolant (550ºC to 800ºC outlet)
  - 25-150 MW_e
  - 15-30 year core life
- **LFR Uses/Benefits**
  - Electricity production, possibly hydrogen production
  - Cartridge core for regional processing
  - Proliferation resistance due to long-life core
- **LFR R&D Challenges**
  - Fuel development
  - High-temperature structural materials
  - Cartridge core/15-20 year refueling



Sodium-Cooled Fast Reactor (SFR)

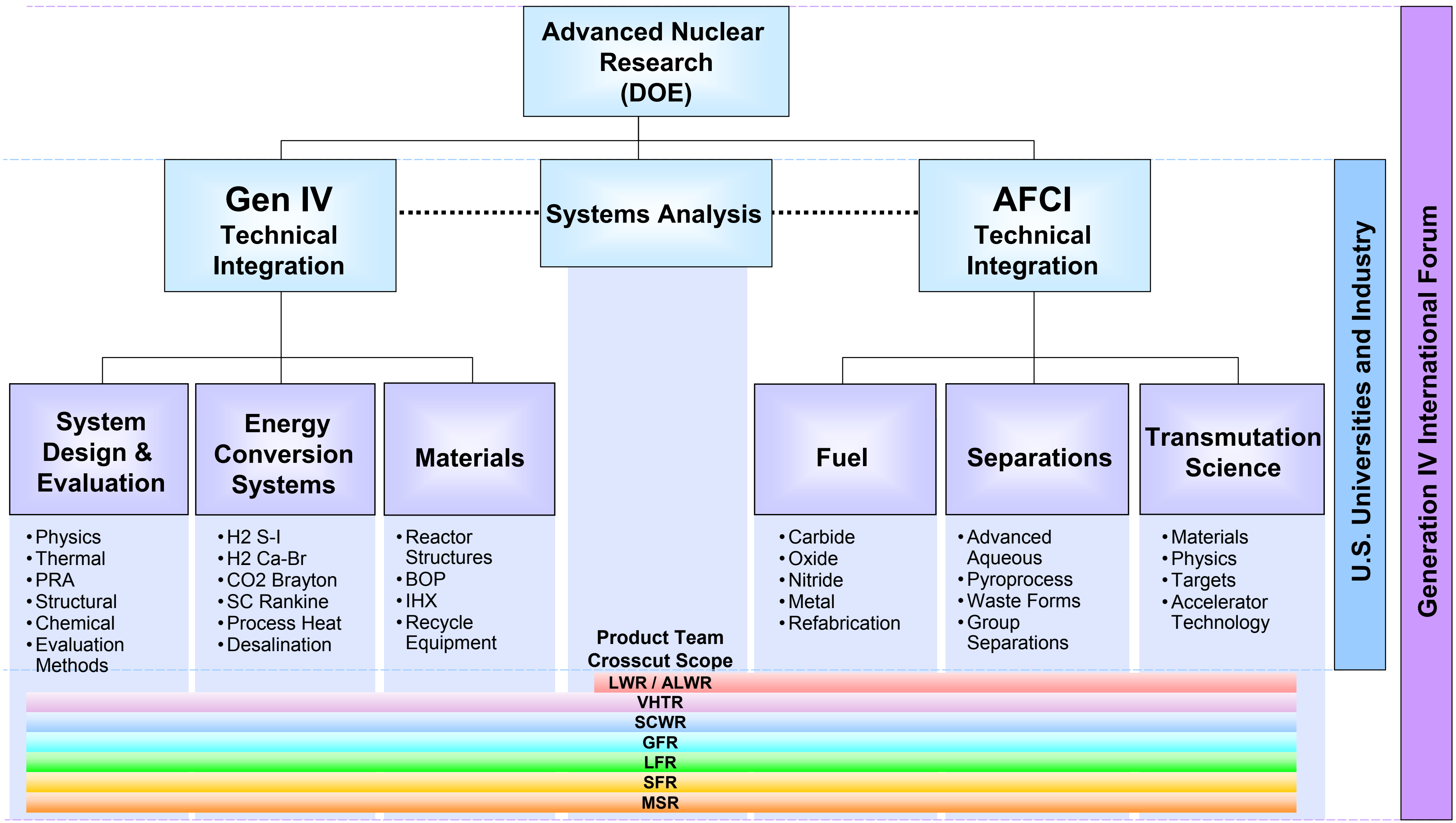
- **SFR Characteristics**
  - Sodium coolant (550 ºC outlet)
  - 150-500 Mw_e
  - Simplified, lower cost design
  - Metal or MOX fuel with advanced recycling technology
- **SFR Uses/Benefits**
  - Electricity production
  - Actinide management
  - Efficient fissile material generation
- **SFR R&D Challenges**
  - Passive safety response
  - Integrate fuel cycle into AFCI
  - Cost reduction



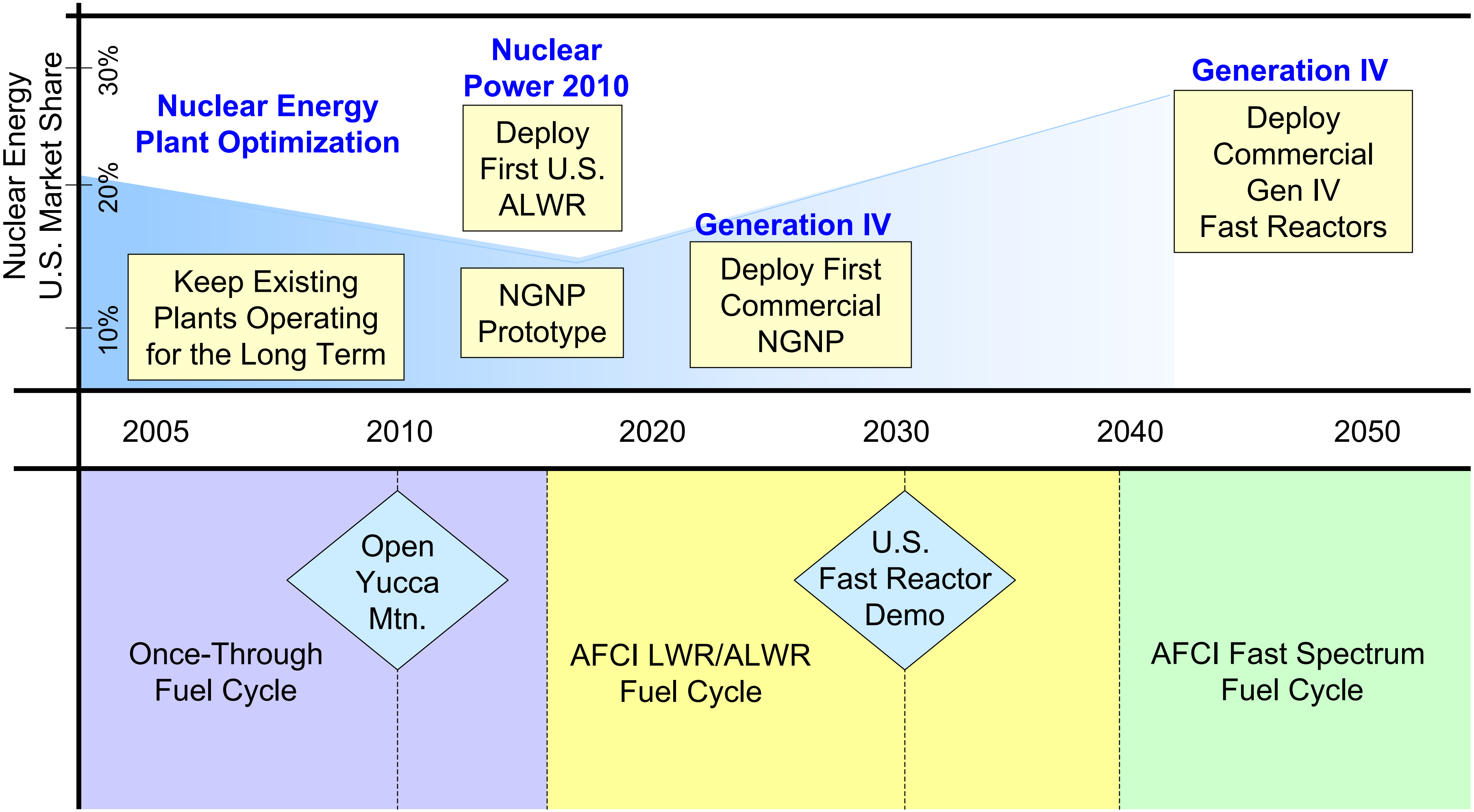
Crosscutting R&D

- **Energy Products**
  - Hydrogen production technology R&D - Sulfur-Iodine cycle, Calcium-Bromine, high-temperature electrolysis
  - Supercritical CO_2 Brayton and supercritical steam Rankine cycle technology R&D
  - Process heat interface R&D
- **Economics**
  - Capital and production cost model
  - Nuclear fuel cycle cost model
  - Energy products model
  - Plant size model
  - Integrated nuclear energy model
- **Proliferation resistance and physical protection (PR&PP)**
  - Evaluation criteria and metrics for PR&PP assessment methodology
  - PR&PP strategy for each reactor design

Gen IV Initiative and AFCI -- an Integrated Program



Long-Term Strategy -- Gen IV and AFCI



Summary

- Generation IV Initiative, in cooperation with GIF, is developing new advanced nuclear systems to realize gains in safety and reliability, economics, sustainability, and proliferation resistance and physical protection
- DOE places first priority on NGNP and SCWR, because it supports President Bush's National Energy Policy, the FreedomCAR Initiative, and the Nuclear Hydrogen Initiative
- DOE places second priority on development of a fast reactor system for waste transmutation and long-term sustainability
- DOE R&D will focus on near-term high-payoff research to support a demonstration facility for the production of hydrogen using nuclear power with longer term R&D supporting advanced reactors that can help close the fuel cycle

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Behavior of Spent Fuel In Dry Casks

Session 4, Chaired by
Ralph Meyer & Charles Interrante of NRC
[Tuesday, October 21, 2003, 11:00 am]

Today’s Presentations

Argonne National Laboratories

- Cladding Behavior During Dry Cask Handling and Storage: H. Tsai, M. Billone
- Mechanical Properties of Irradiated Zr-4 for Dry Cask Storage Conditions and Accidents: R. Daum, S. Majumdar, M. Billone

Current Information Needs

Background, ongoing and planned work

- **P** License Renewal Activities
  - Gave rise to “User Needs” related mainly to creep data
  - EPRI, DOE, NRC coordinated effort at INEEL, ANL
  - Surry’s Renewal Application is Already in Progress.
- **P** Cask Behavior in Licensed Service
  - As Burnup levels increase, needs arise for improved modeling that requires better understanding of selected mechanical properties & fracture toughness data. Licensing activities require guidance for meeting the safety functions of the storage system: thermal, radiological, confinement, sub-criticality, retrievability.

License Renewal

Recent pertinent reports

- NUREG/CR-6831
  - "Examination of Spent PWR Fuel Rods after 15 Years in Dry Storage," [Einziger, et al., 2003]
    - Residual Creep Capacity of Surry Fuel taken from INEEL's Castor V/21 is shown to be adequate for the storage, transport and disposal parts of the fuel cycle.
- ASTM Standard Guide "C 1562 - 03"
  - "Evaluation of Materials Used in Extended Service of Interim SNF Dry Storage Systems"
    - [ASTM TG under C26.13 on Spent Fuel and High Level Waste]
    - This guide followed an NRC sponsored study by SAIC on "Technical Basis for License Renewals for ISFSI."

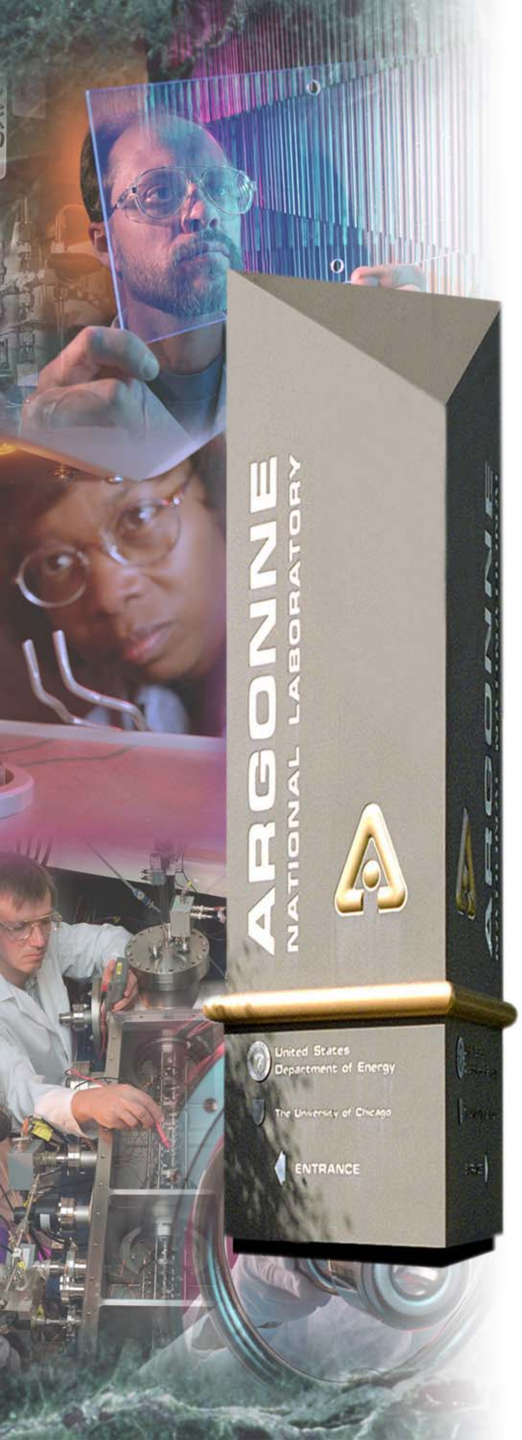
Cask Behavior in Licensed Service

- Mechanical Properties & Fracture Toughness of a Fuel Assembly
  - Model the behavior over pertinent temperatures for fuel assemblies using estimates of condition and properties along the lengths of the each rod and within each assembly.
- Cladding Behavior Models for SNF Assembly for abusive and accident service conditions
  - Fracture Modeling
    - Fuel cladding condition (oxides & hydrides along the length of the rods)
    - Fracture by brittle or ductile mode: fracture toughness, impact strength, and stress / strain to fracture are needed.
  - Dispersion Modeling of Fuel
    - Pellet fracture properties as function of service conditions: e.g. higher burnup levels lead to fine-grained structure in the rim of the pellet

Cask Behavior in Licensed Service

Continued

- Hydrides in Zirconium based Clad Materials
  - Circumferential hydrides form in reactor service and their volume fraction increase with H content.
  - Radial hydrides seem to form from supersaturated solution at stress levels above ~90 MPa.
  - Thermal cycling at high tensile stress promotes radial hydride precipitation.
  - Embrittlement increases potential for longitudinal fracture of rods.



Cladding Behavior during Dry Cask Handling and Storage

Hanchung Tsai (htsai@anl.gov)

Nuclear Safety Research Conference
October 20-22, 2003
Washington DC

Argonne National Laboratory

Office of Science
U.S. Department of Energy

A U.S. Department of Energy
Office of Science Laboratory
Operated by The University of Chicago

Cladding Behavior during Dry Cask Handling and Storage

- **Maintaining SNF cladding integrity is important for cask performance**
  - Fuel retrievability, cask surface dose rate, criticality.
- **Factors that may affect cladding behavior include**
  - Thermal creep
  - Hydrogen in cladding – hydride redistribution under the influence of temperature and stress.
- **Objective of our work is to provide data to support**
  - Extending storage time (>20 years), and
  - Extending burnup (>45 GWd/MTU).

Cladding Behavior during Dry Cask Handling and Storage

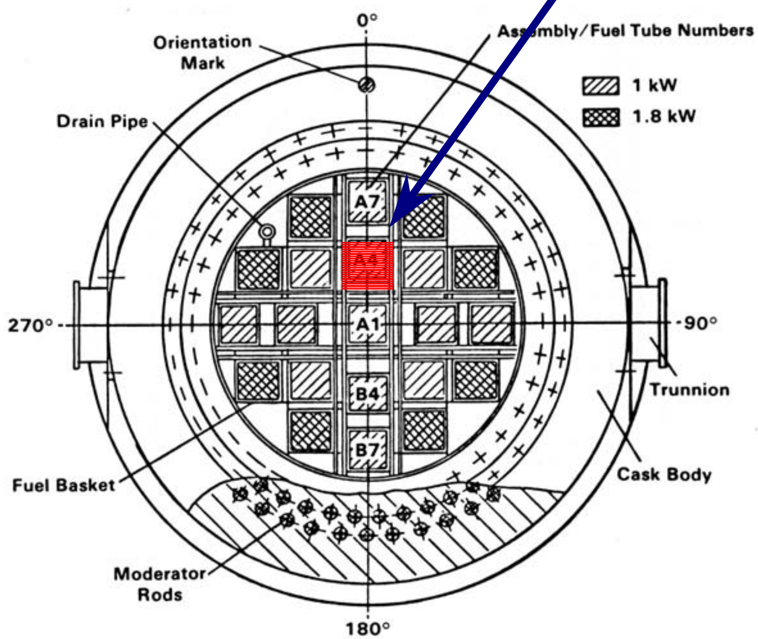
We have performed the following:

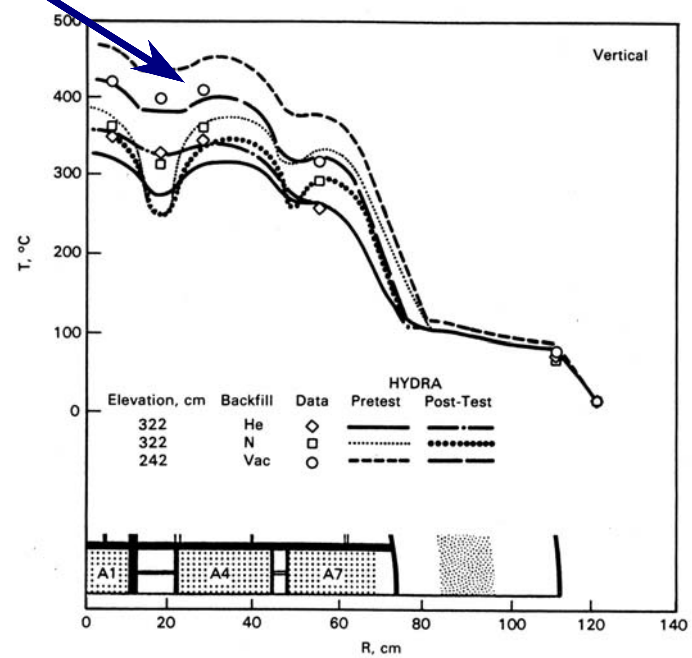
- Characterization of medium-burnup (36 GWd/MTU) Surry PWR rods after 15 y-storage in a Castor-V/21 dry cask
  - Extensive in-cask thermal benchmark tests, some emulated vacuum drying.
- Isothermal annealing of cladding from high-burnup (67 GWd/MTU) H. B. Robinson rods
  - Conditions relevant to vacuum drying (420-500°C, 2-72 h).
  - Post-annealing microhardness and hydride morphology determinations.
- Thermal creep tests of both Surry and Robinson cladding
  - Creep ductility and hydride reorientation.

Effects of 15-y Dry Cask Storage (and thermal benchmark tests) on Surry Rods

Surry Rods (Assembly T11) in Castor-V/21

Assembly T11



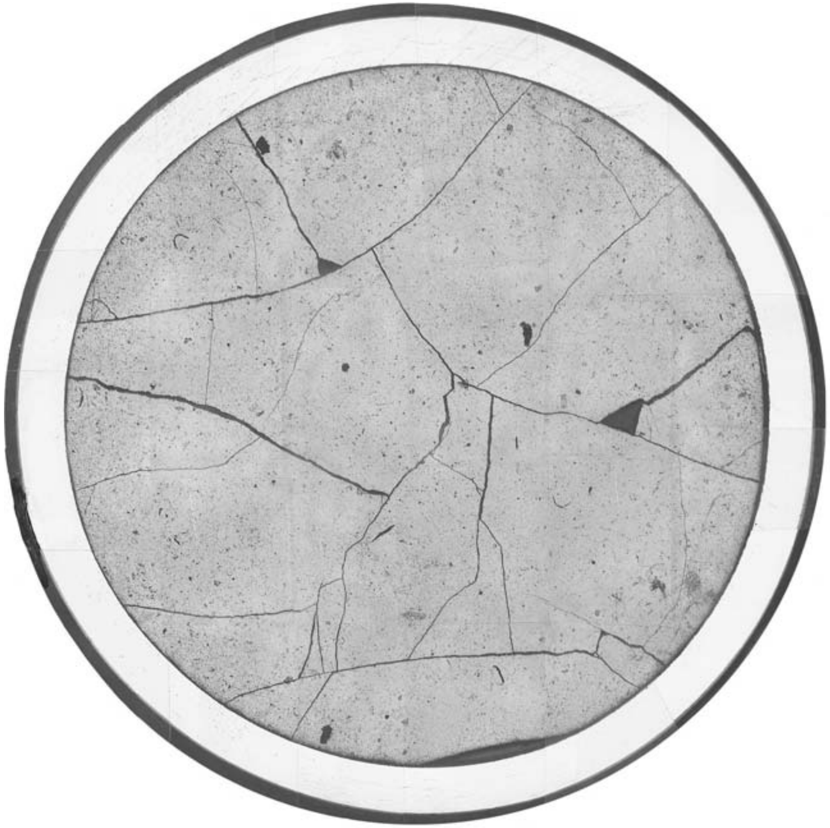


Location of source rods in the Caster-V/21 cask

Peak cladding temperature ≈415°C for 3 days when the cask was in vacuum. Cladding hoop stress was, however, low, <70 MPa.

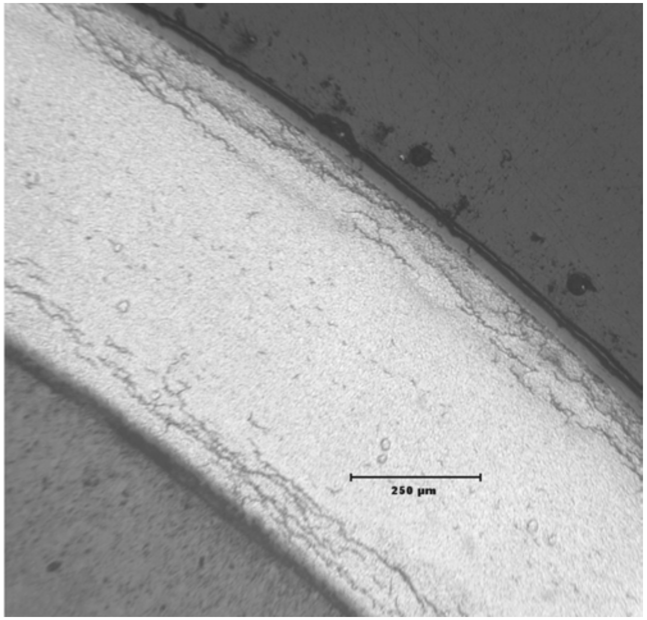
Surry Post-Storage Characterization

- **Effect of 15-y storage is benign**
  - **Gas release: ≈0.5-1.0 %**
    - **No additional release**
  - **Fuel microstructure**
    - **No obvious changes**
  - **$\Delta D/D_{as\text{-}built}$: ≈-0.6%**
    - **Little or no in-storage creep**



Surry Post-Storage Characterization *(cont'd)*

- **Effect of 15-y storage is benign.**
  - Cladding microhardness: 235-240 DPH
    - No apparent annealing in storage
  - OD oxide thickness
    - Normal (≈24-33 μm)
  - Cladding hydrogen content
    - Normal (≈250-300 wppm)
    - Axial migration - tbd
  - Hydride reorientation
    - None observed



Surry Post-Storage Characterization *(cont'd)*

- **Summary**
  - 15-y dry-cask storage (with extensive in-cask thermal benchmark tests) produced no apparent deleterious effects on the Surry rods.
  - Segments of Surry cladding were prepared for post-storage thermal creep and tensile tests.

Cladding Annealing Tests and Hydride Reorientation

Robinson Cladding Annealing Tests

- **During vacuum drying, cladding temperature may be raised to >≈400°C for hours to days. Will this alleviate radiation hardening in the cladding? What effect it has on hydrogen distribution?**
  - **Figure of merit: cladding microhardness**
  - **Test samples: short segments of defueled cladding from center of rod (11.3×10^{21} n/cm^2, E>1 MeV, ≈600 wppm H)**
  - **Corollary objective: study hydride redistribution under stress-free conditions**
  - **Test environment: high-purity argon**

Robinson Cladding Annealing Tests

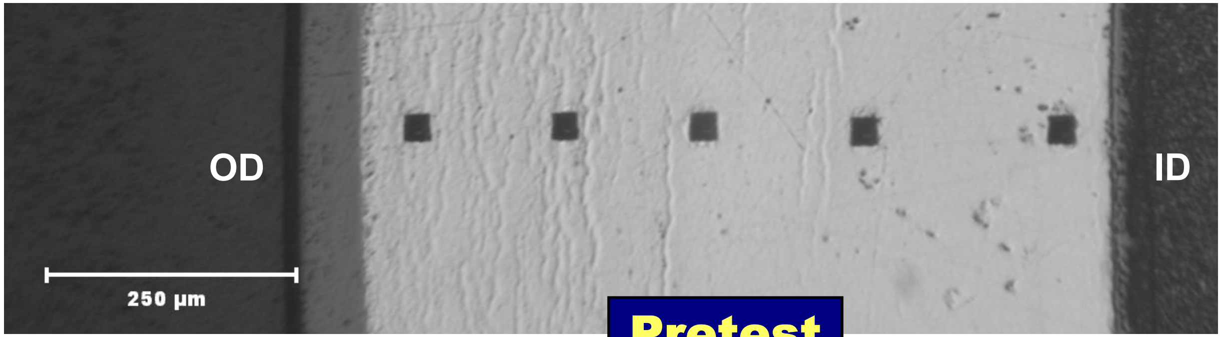
- **Annealing Test Matrix**

	2 h	10 h	20 h	48 h	72 h
420°C			C6		C7
450°C	C8	C9			
500°C	C10			C11	

Robinson Cladding Annealing Tests

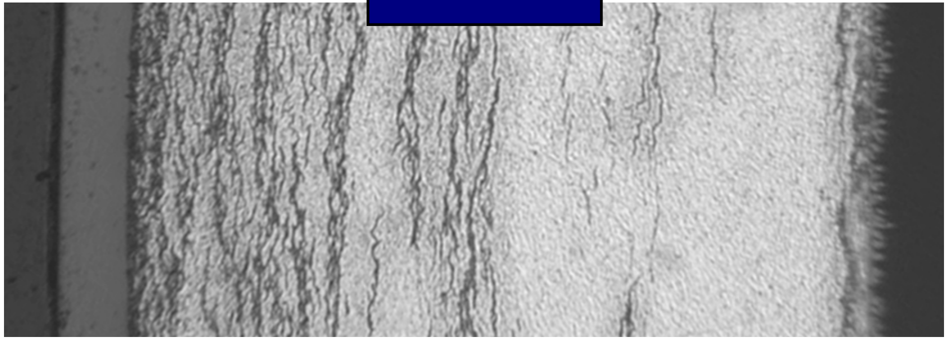
- **Microhardness Determination**
  - Apply a known load with a diamond tip, measure the size of the indentation, and convert to microhardness (DPH)

DPH uniform across thickness, Avg = 252



Pretest

Etched, showing hydrides ≈600 wppm H



Robinson Cladding Annealing Tests

- **Microhardness Determination**
  - For nonirradiated sibling: Ho = 203
  - For as-irradiated sibling: Hi = 252

Microhardness after annealing tests

	2 h	10 h	20 h	48 h	72 h
420°C			226		215
450°C	224	217			
500°C	218			206	

Robinson Cladding Annealing Tests

$$\text{Recovery} = \left[1 - \frac{\text{H} - \text{Ho}}{\text{Hi} - \text{Ho}}\right]$$

% Radiation Hardening Recovery

	2 h	10 h	20 h	48 h	72 h
420°C			54		75
450°C	58	71			
500°C	69			94	

Conclusion: Given time, significant recovery will occur at T >≈420°C.

Robinson Cladding Annealing Tests

- **Hydride Morphology Evolution**
  - Strongly governed by hydrogen solubility in Zircaloy

Temperature (°C)	Solubility (wppm)
25	0
200	13
400	200
420	240
450	310
500	460

Surry: 300 wppm

Robinson: 600 wppm

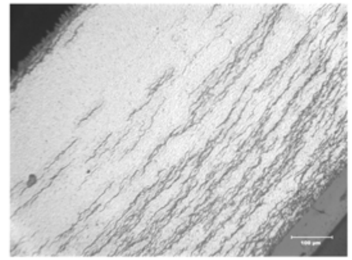
J. J. Kearns

Robinson Cladding Annealing Tests

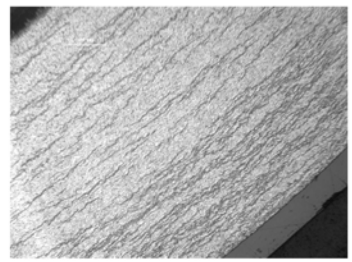
- **Hydride Morphology Evolution**
  - **Precipitates became more uniformly distributed across the thickness**
    - ***Temperature. time***
  - **No radial reorientation (being stress-free)**

Hydride Morphology

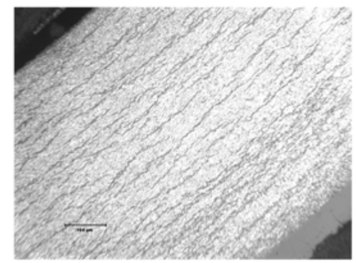
H. B. Robinson Cladding Annealing Test Samples



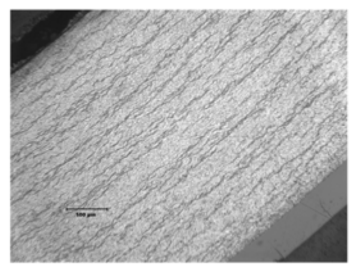
611C2 As-irradiated Control



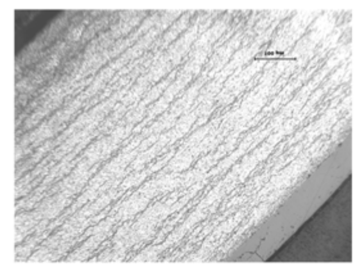
611C6 420°C, 20 h



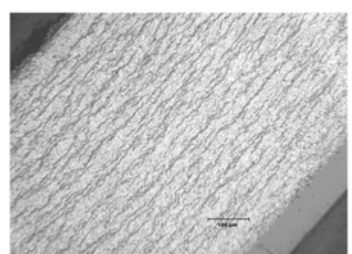
611C7 420°C, 72 h



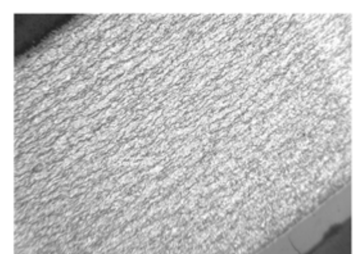
611C8 450°C, 2 h



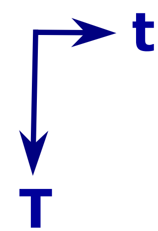
611C9 450°C, 10 h



611C10 500°C, 2 h

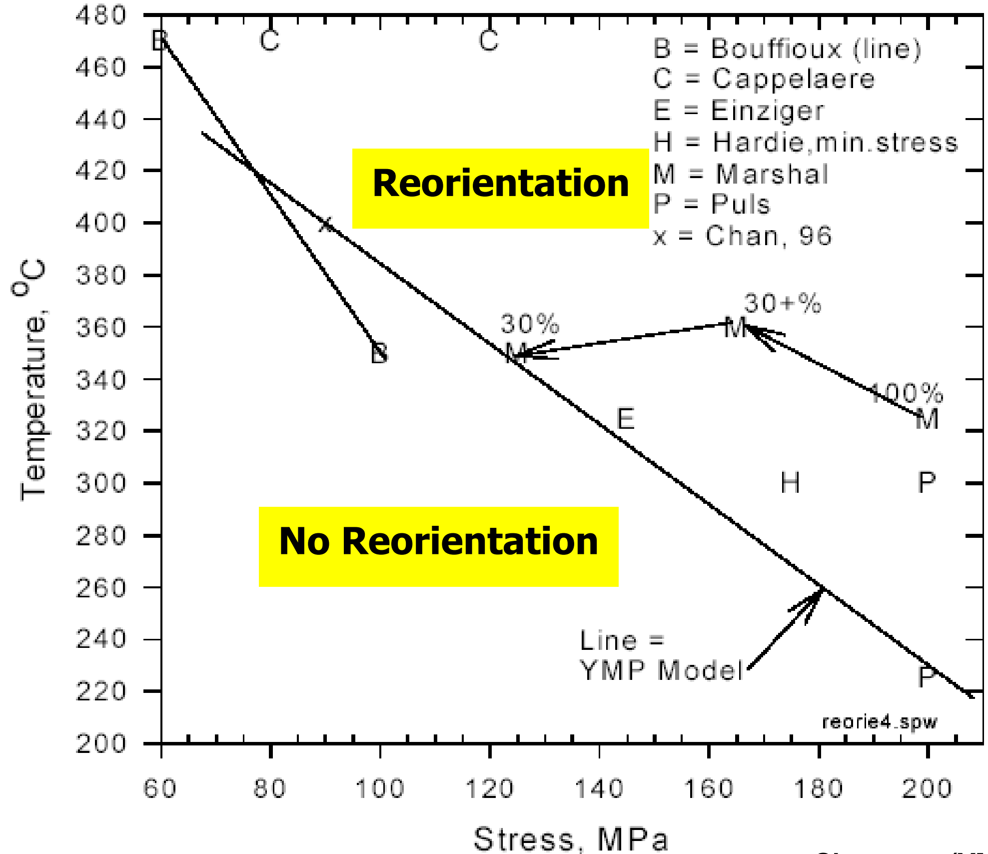


611C11 500°C, 48 h



Hydride Reorientation – Creep Tests

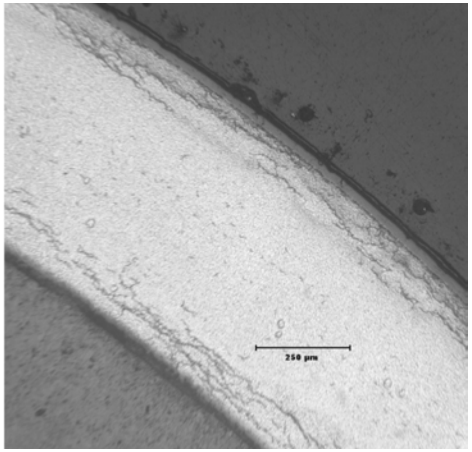
- **Radial hydrides, as little as 40 wppm, can significantly degrade cladding's mechanical properties.** **(Marshall)**
- **Stress, temperature, cool-down rate, microstructure, H content, etc., all play important roles.** **(Einziger)**
  - **Threshold hoop stress for 400°C is ≈100 MPa.**



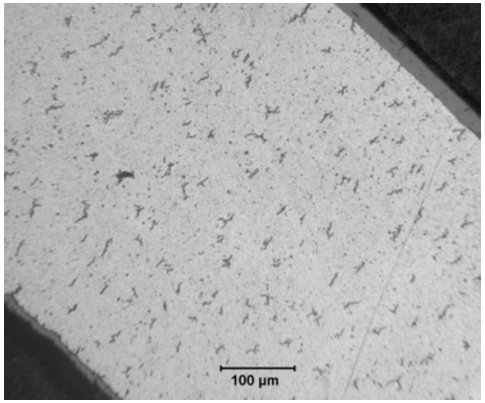
Siegmann (YMP)

Hydride Reorientation – Creep Tests

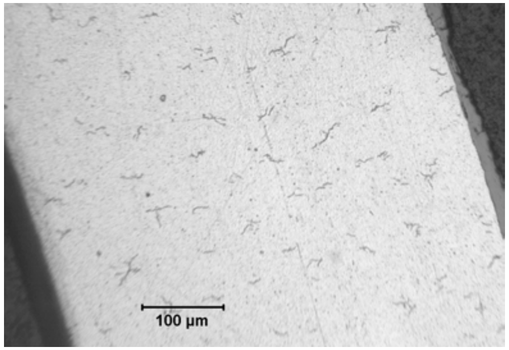
- Two Surry creep tests were intentionally shut down with samples under pressure: C3 (360°C, 220 MPa, 3760 h, 0.22% ε) and C6 (380°C, 190 MPa, 2400 h, 0.35% ε).
- Both samples survived the shutdown.
- Hydrides redistributed. Some now in radial direction, but no long-range linkage



Pretest



Posttest C3



Posttest C6

Hydride Reorientation – Creep Tests

CEA (Cappelaere et al, ICEM 2001) – 470°C

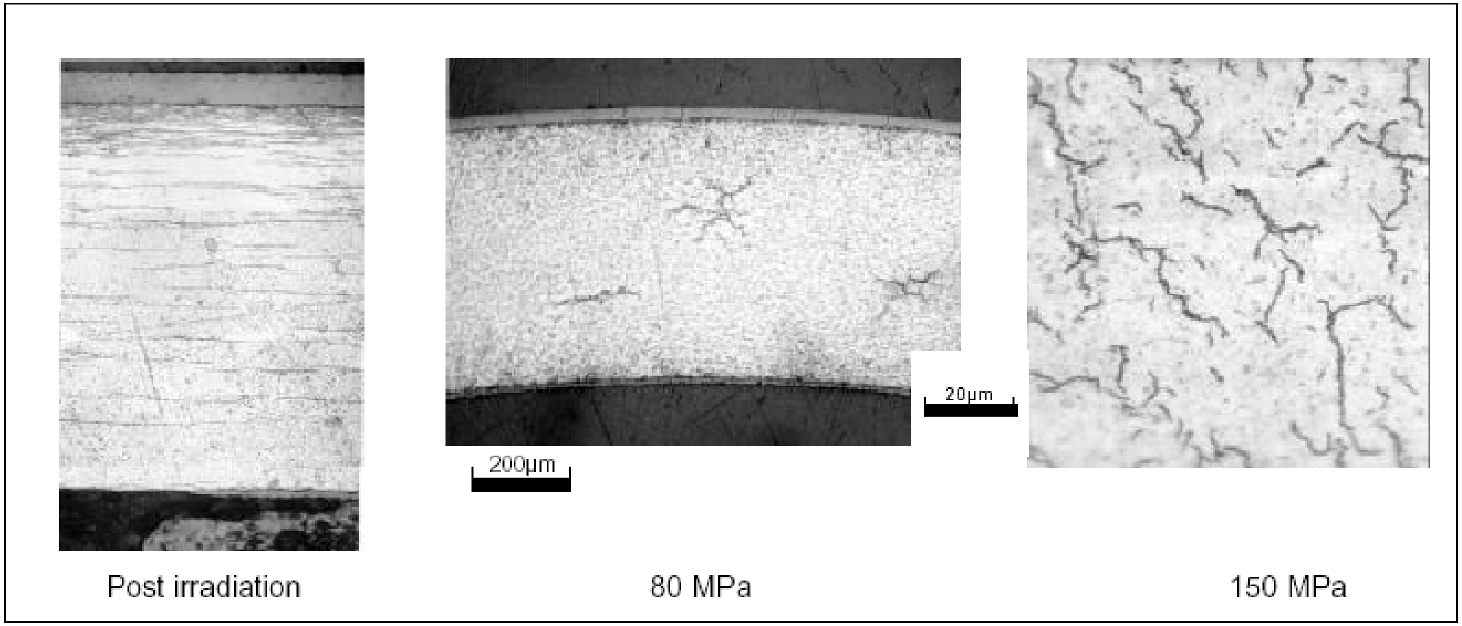
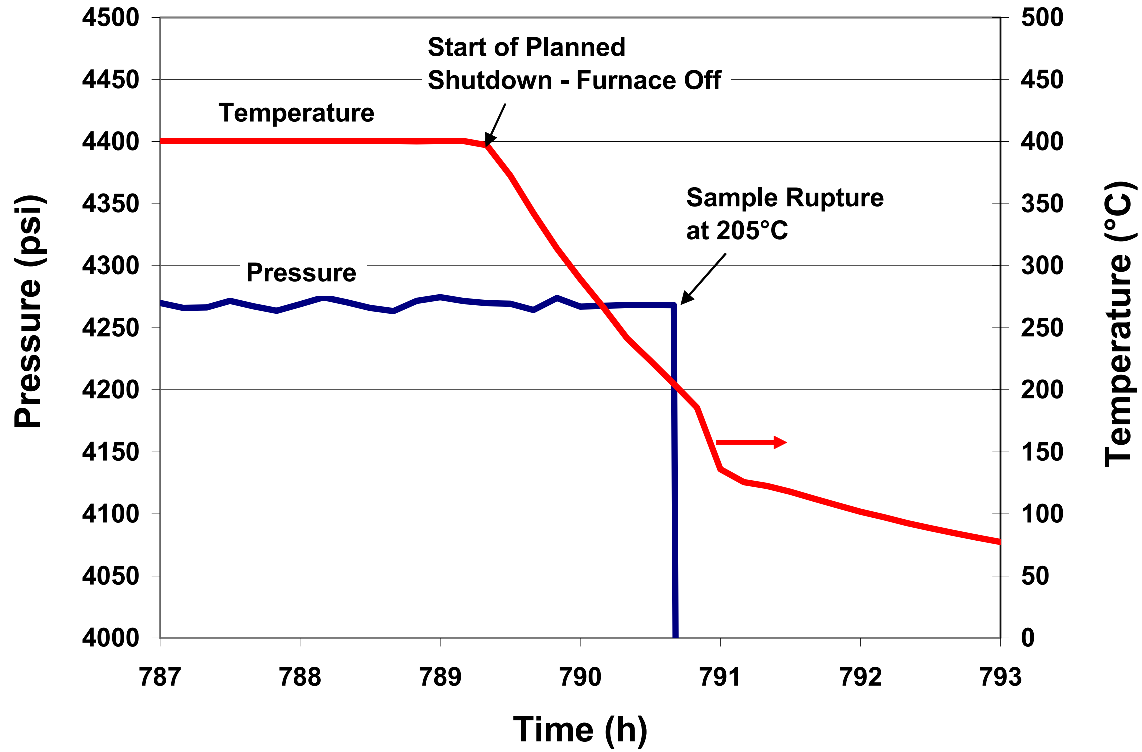


Figure 6 : Impact of creep tests on hydrides morphology, distribution and orientation

Hydride Reorientation – Creep Tests

- **Shutdown-under-pressure was repeated for one of the high-burnup H. B. Robinson creep sample: C15 (400°C, 190 MPa hoop, 2440 h, ≈3.5% ε).**

- **The sample ruptured during cool-down at 205°C.**
- **Cause being investigated**
  - **Hydride Reorientation?**



Cladding Thermal Creep

Thermal Creep Tests

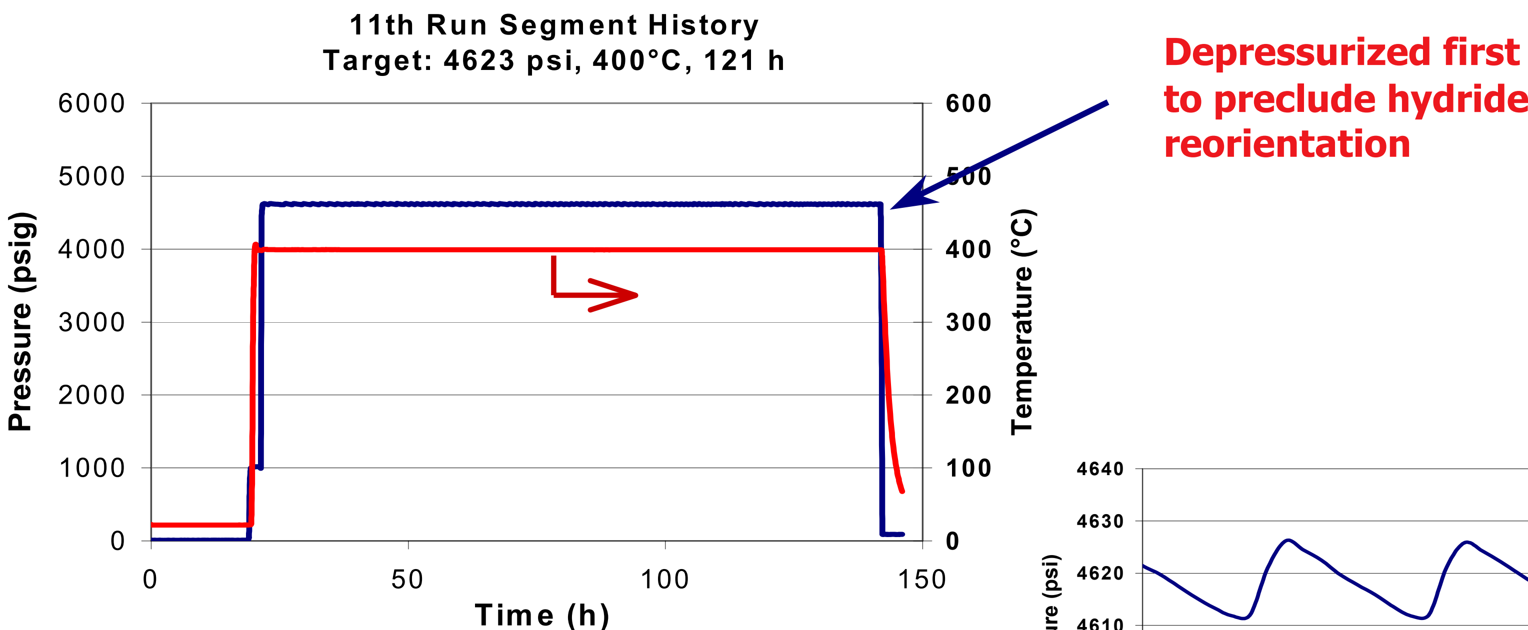
- **Why studying creep?**
  - **Creep is the dominant cladding deformation mechanism under normal conditions of dry storage. The core issue is, of course, cladding integrity.**
- **Test objectives**
  - **Determining steady-state creep rate and ductility limit.**
  - **Generating samples to study hydride reorientation and post-creep mechanical properties.**

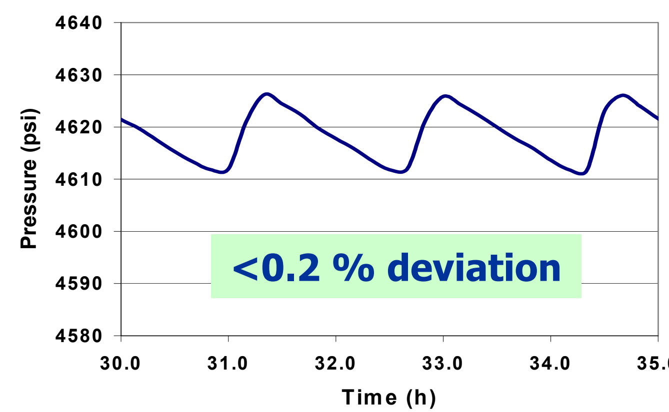
Thermal Creep Tests

- Creep Test Specimen
  - 76-mm-long segments of irradiated cladding.
  - Welded end fittings.
  - Pressure actively regulated.

Thermal Creep Tests –Typical Performance

- **Good pressure and temperature control**
- **Periodic shutdowns for laser profilometry**

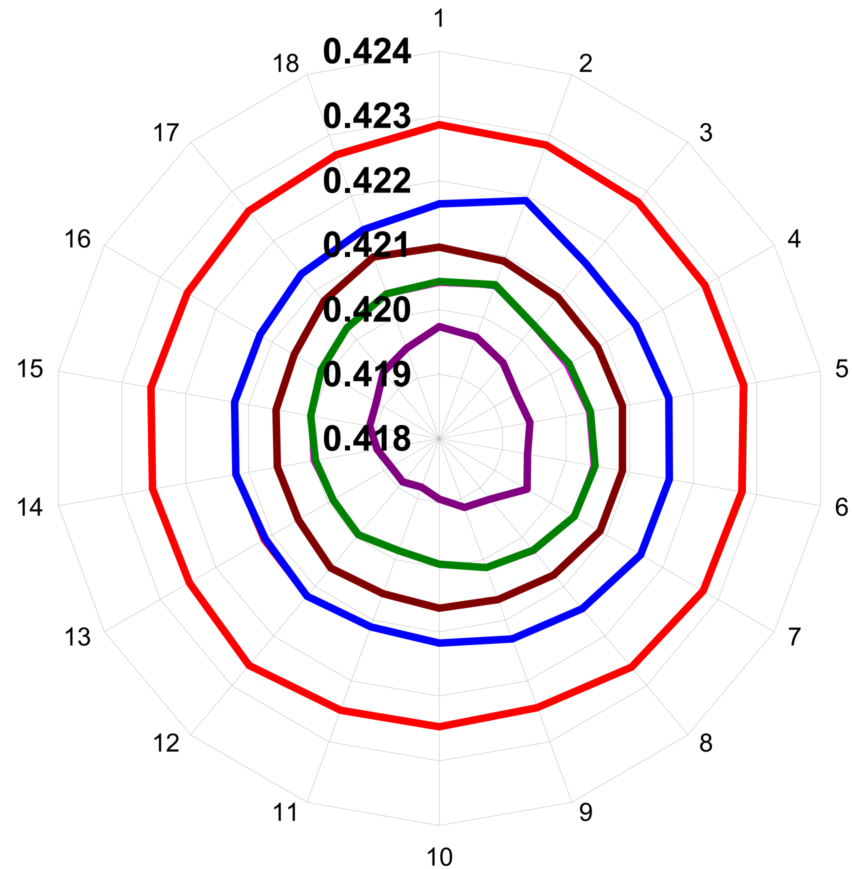




Thermal Creep Tests

Laser Profilometry – Typical Results

- Midplane cross-sectional profiles of a sample at 0, 335, 671, 1028, and 1820 h. (Dimensions in inches.)



Thermal Creep of Post-Storage Surry Cladding

Surry Summary Results

Sample	Temp. (°C)	Stress (MPa)	At End of Test			Sample Disposition
			Hours	Avg. ε	Intact?	
C3	360	220	3305	0.22	Yes	DE(1)
C6	380	190	2348	0.35	Yes	DE(1)
C8	380	220	2180	1.10	Yes	Bend Test
C9	400	190 250	1873 693(2)	1.03 5.83	Yes Yes	-- Bend Test
2-C9	400	160	286(3)	0.22	Yes	tbd

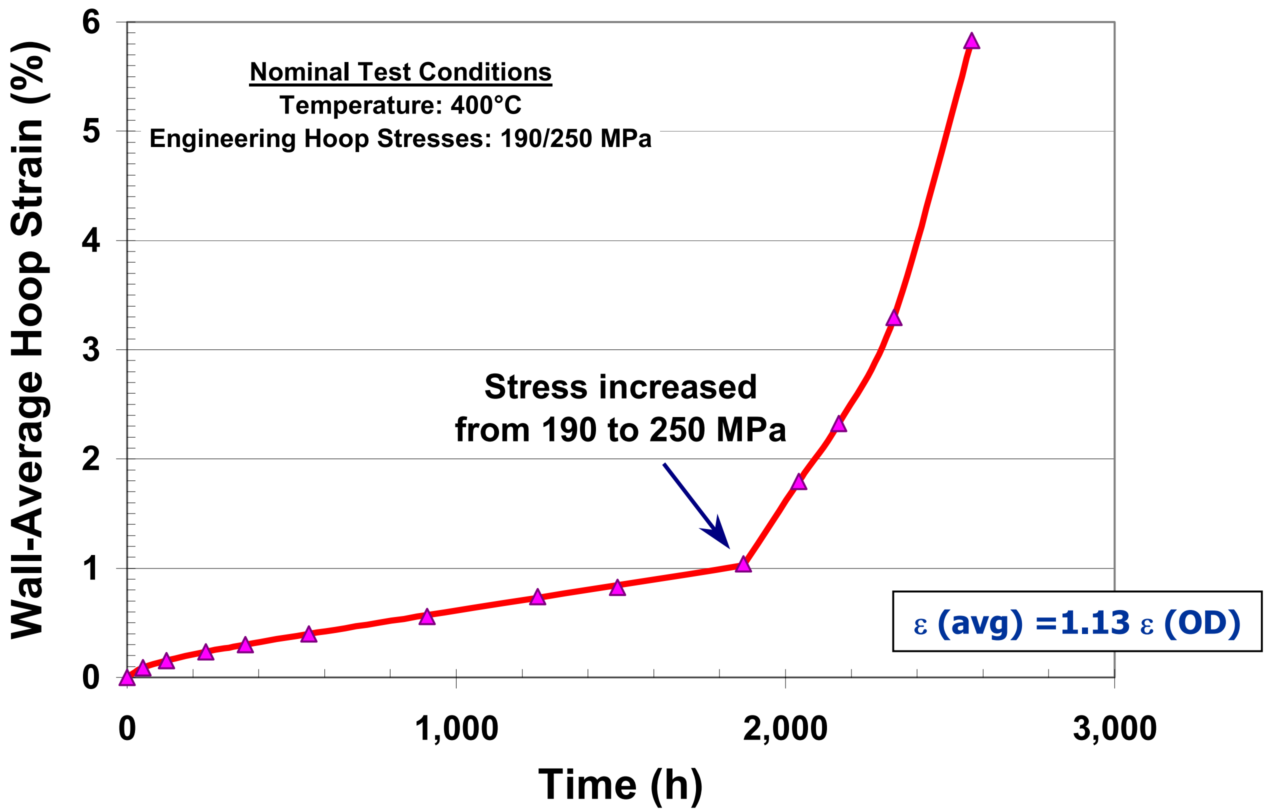
(1) DE: Destructive examination, for hydride orientation determination. For this, the final shutdown was done with sample pressurized.

(2) Incremental hours

(3) On-going

Thermal Creep Tests – Surry C9

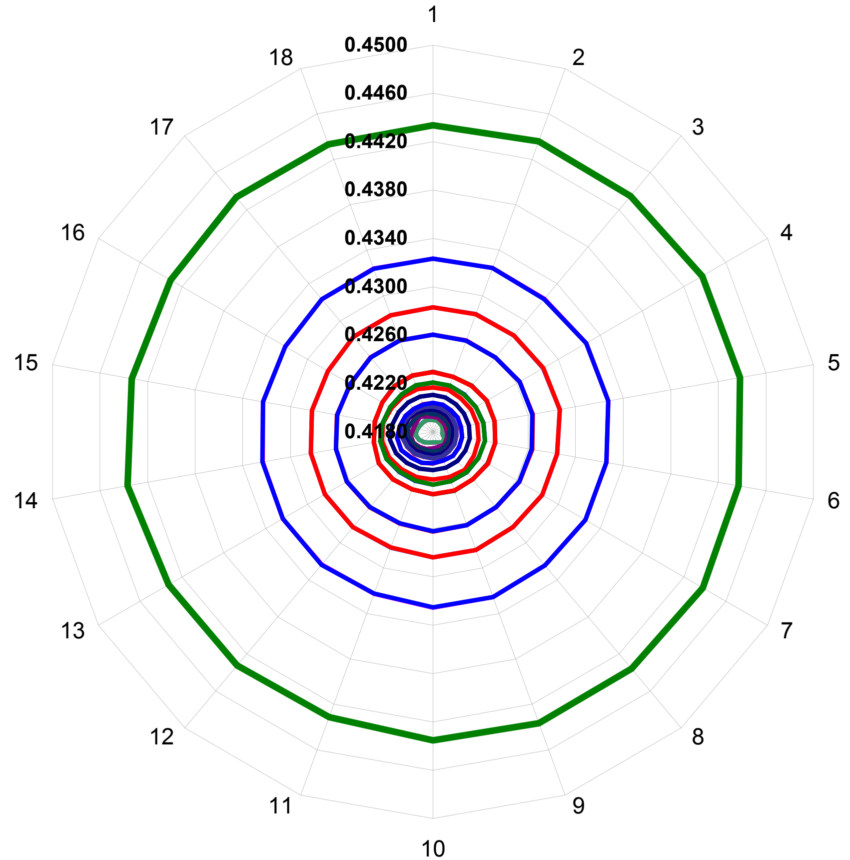
- 400°C, 190/250 MPa engineering hoop stress, 2566 h
- 5.8% average hoop strain, no rupture



Thermal Creep Tests – Surry C9

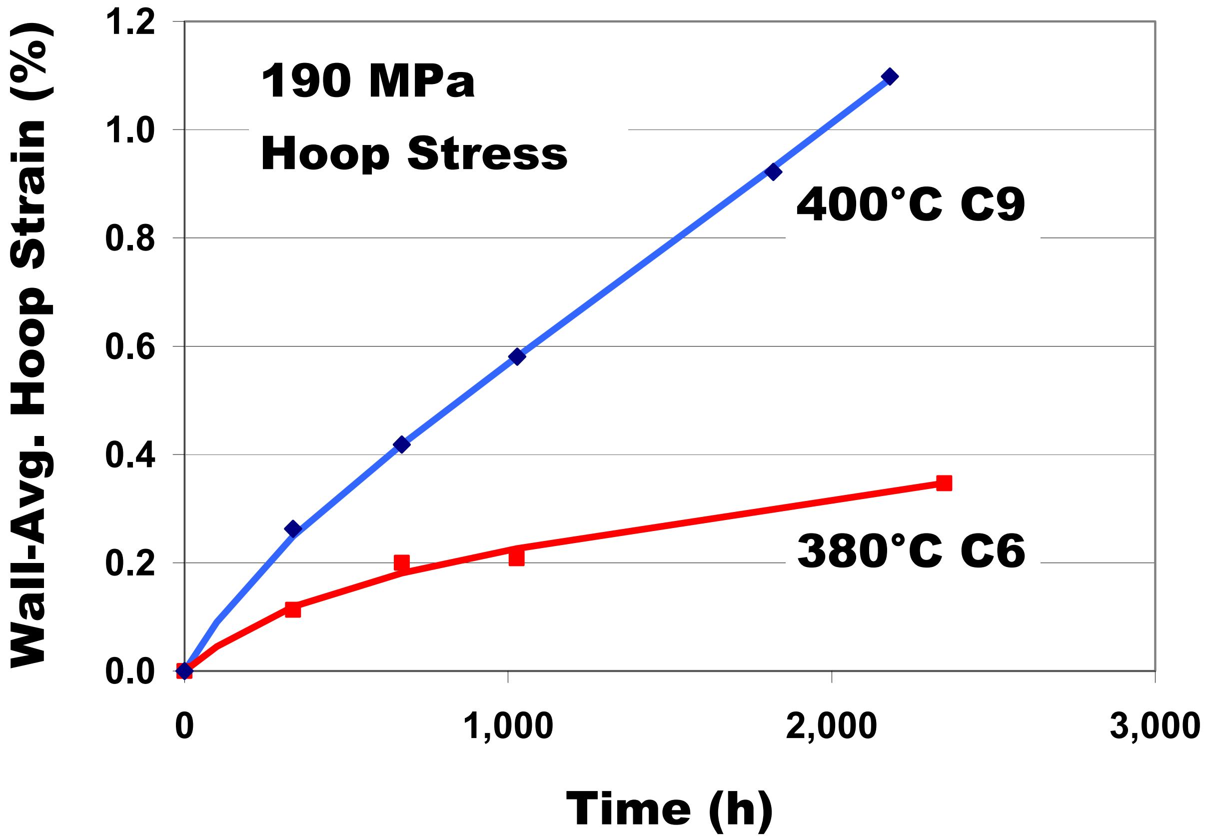
- **Deformation uniform even at high strain (5.8%)**
- **No signs of imminent failure**
- **Additional creep ductility likely**

Run-by-Run Cross Sectional Profiles of C9 (Dimension in inches)



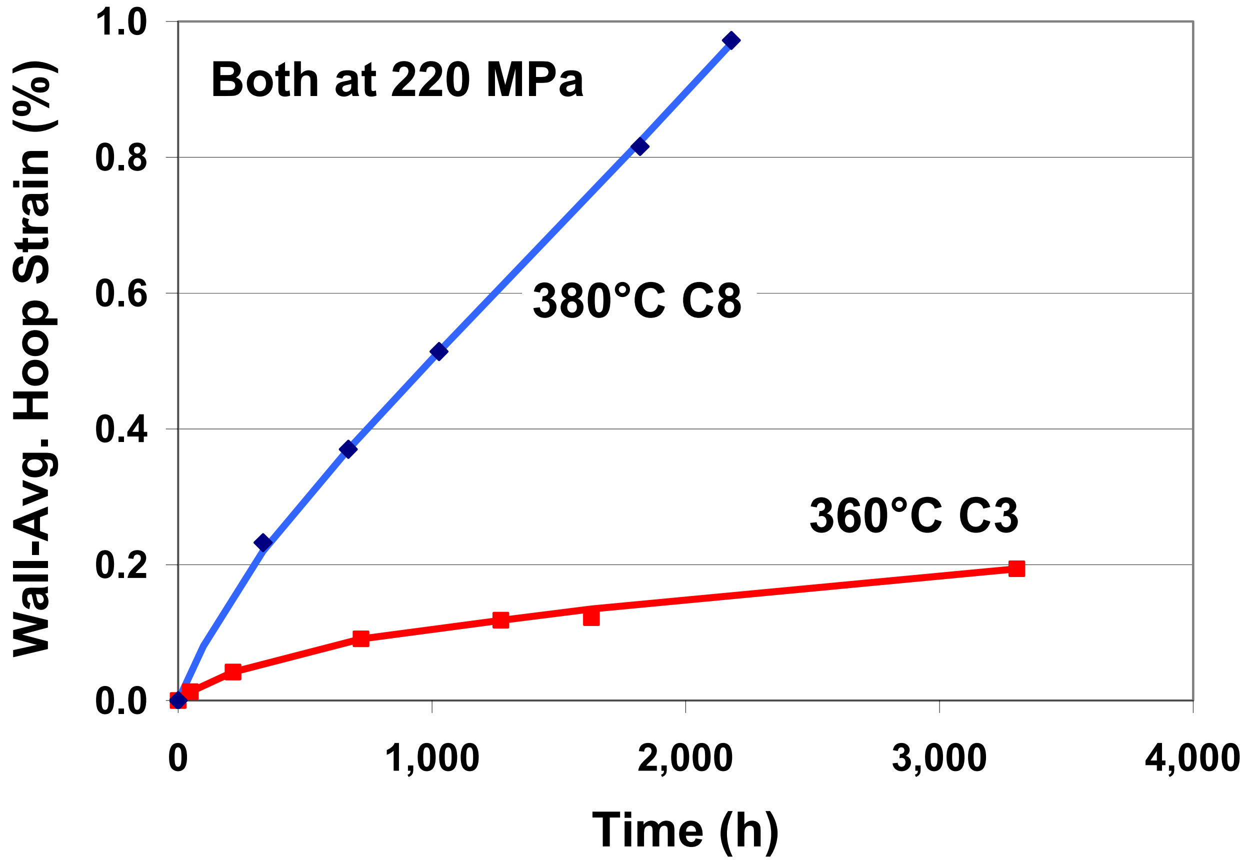
Thermal Creep of Post-Storage Surry Cladding

- Temperature Dependency



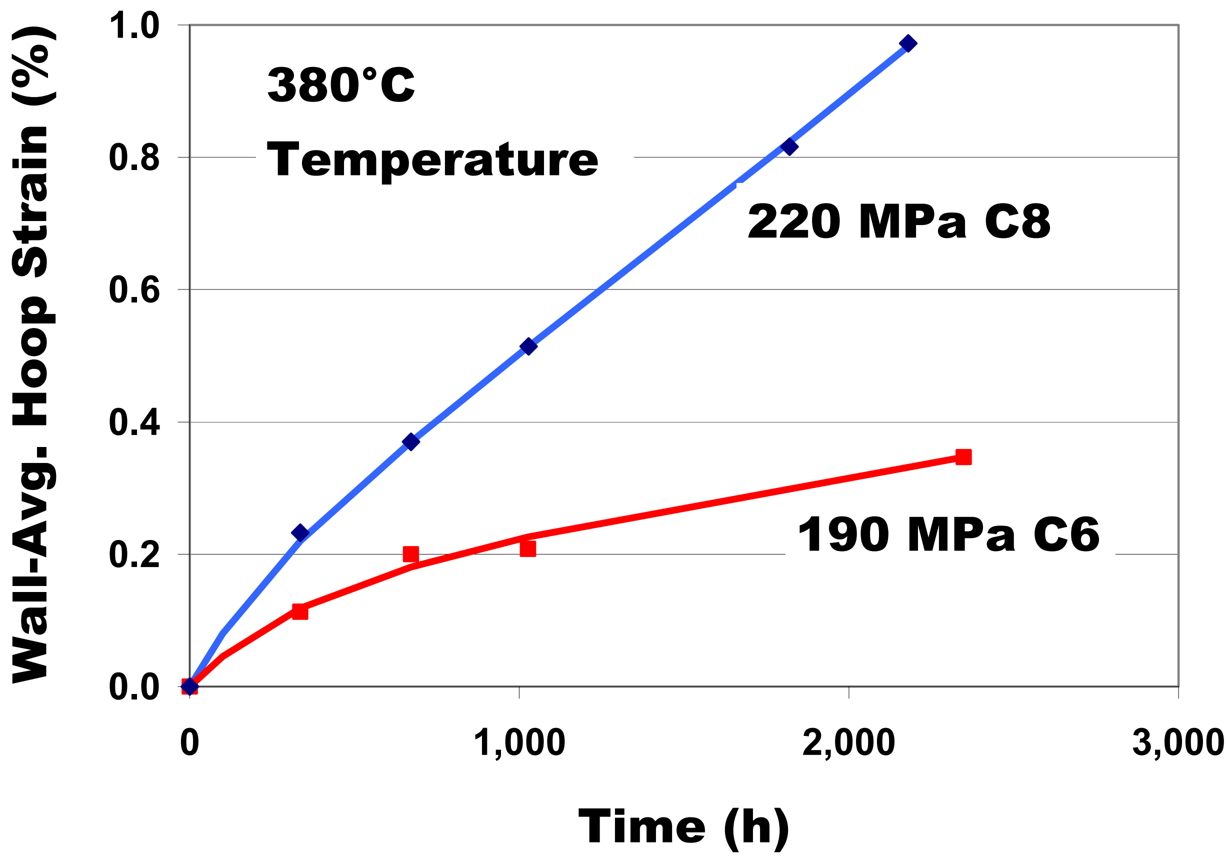
Thermal Creep of Post-Storage Surry Cladding

- **Temperature Dependency**



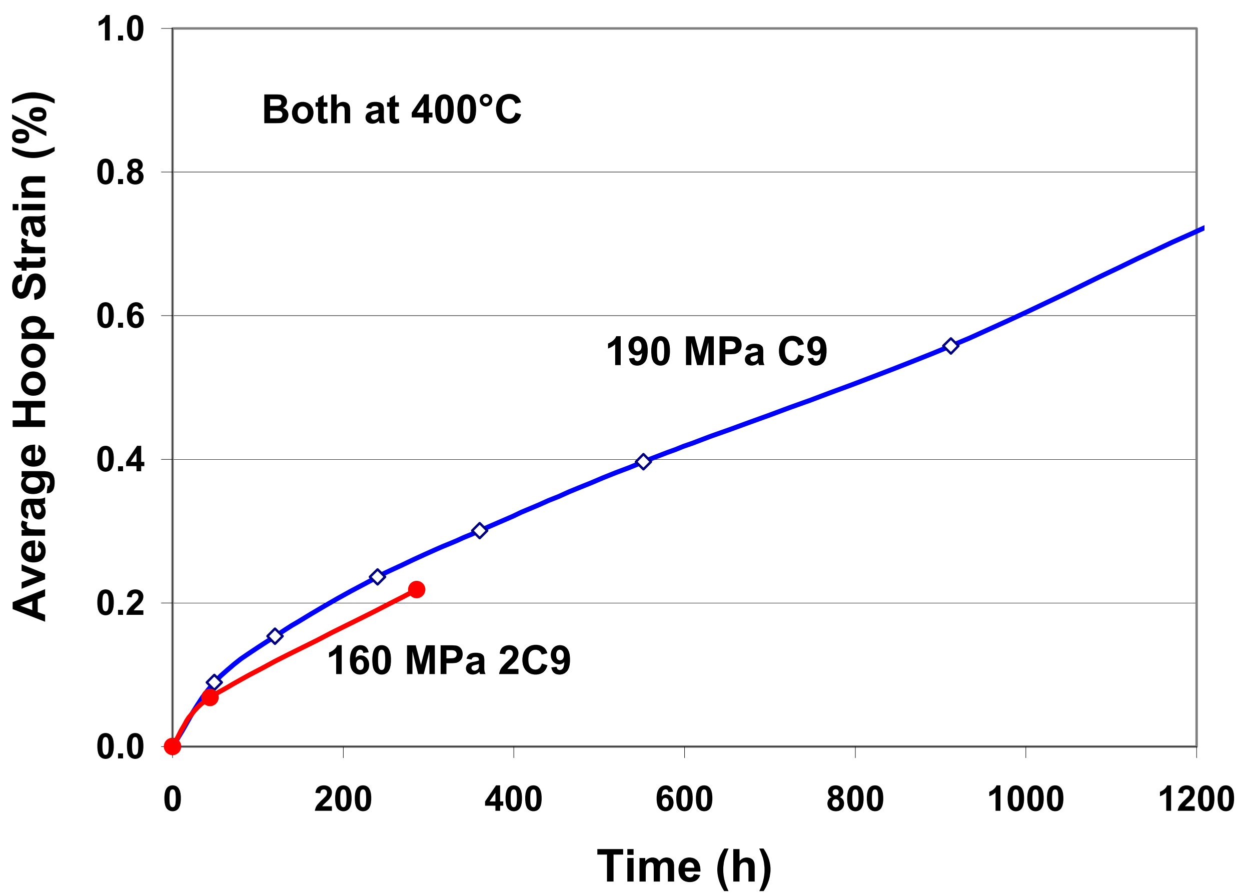
Thermal Creep of Post-Storage Surry Cladding

- **Stress Dependency at 380°C**



Thermal Creep of Post-Storage Surry Cladding

- **Stress Dependency at 400°C**



Thermal Creep of Post-Storage Surry Cladding

Secondary Creep Rates

Test Purpose	Sample	Temp. (°C)	Stress (MPa)	SS $\Delta\varepsilon/\Delta t$[1] (%/h)
PSC	C3	360	220	$\approx 1.6\times10^{-5}$
PSC	C6	380	190	$\approx 8.6\times10^{-5}$
RCS	C8	380	220	$\approx 4.6\times10^{-4}$
RCS	C9	400	190 250	$\approx 4.9\times10^{-4}$ $\approx 4.9\times10^{-3}$

(1) ε (avg). Values are approximates. Effects of wall thinning and diameter increase on hoop stress not included.

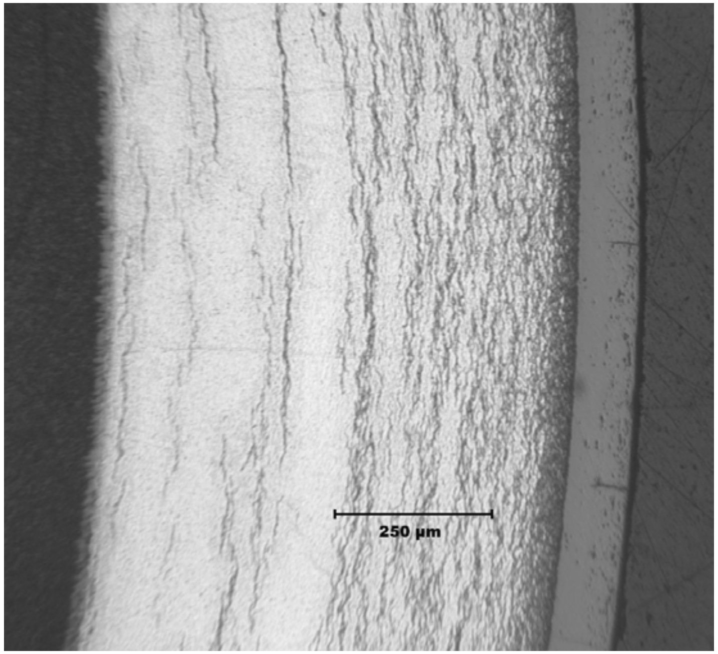
Thermal Creep Tests – H. B. Robinson

Robinson Test Matrix

		Stress (MPa)				
		100	160	190	220	250
Temp. (°C)	420		1			
	400		1	C14 C15	1	
	380		1	C16	C17	
	360			1	1	
	320					

H. B. Robinson Cladding

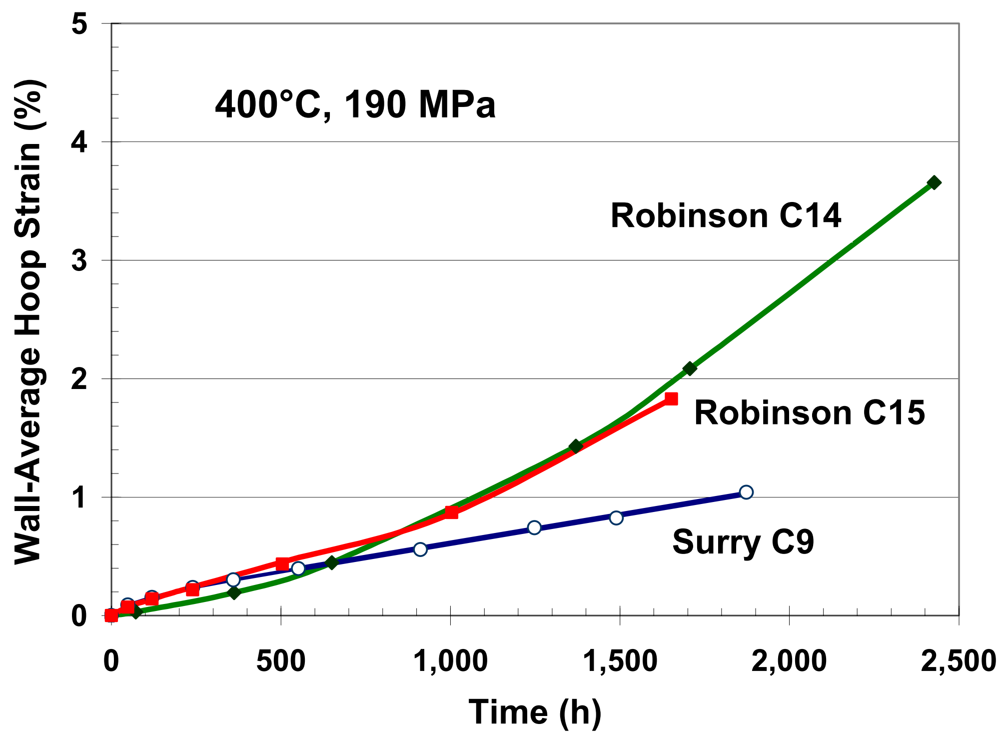
- **Significant corrosion and H uptake from extended operation to high burnup**
  - *≈100 μm max. oxide*
  - *≈800 wppm max. hydrogen*
  - *Circumferentially-oriented hydrides*



- **What are the effects of increased hydrogen and radiation damage on creep?**

Thermal Creep Tests – H. B. Robinson

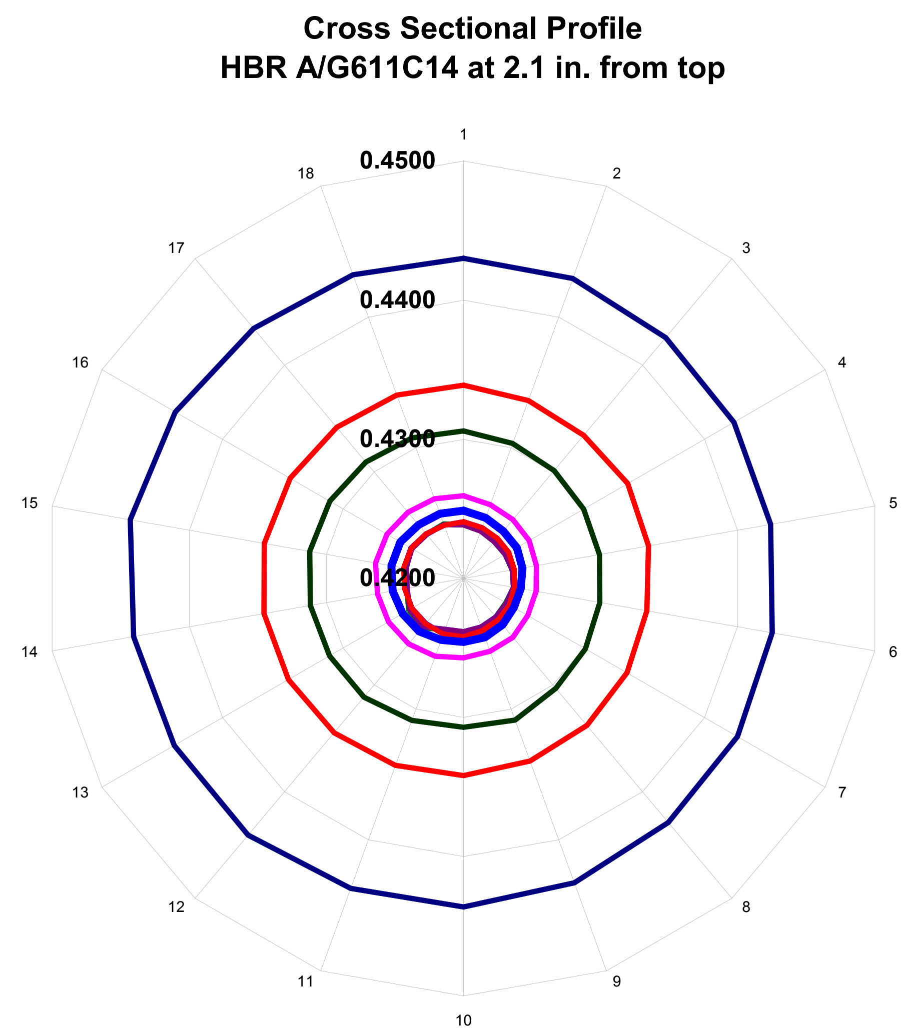
- At 400°C, creep rate of H. B. Robinson appears to be comparable to that of Surry at the onset of test. Rate becomes greater afterwards, possibly due to annealing.
- C14 was terminated at 2450 h at 3.6% ε. Sample was intact.



Thermal Creep Tests – H. B. Robinson

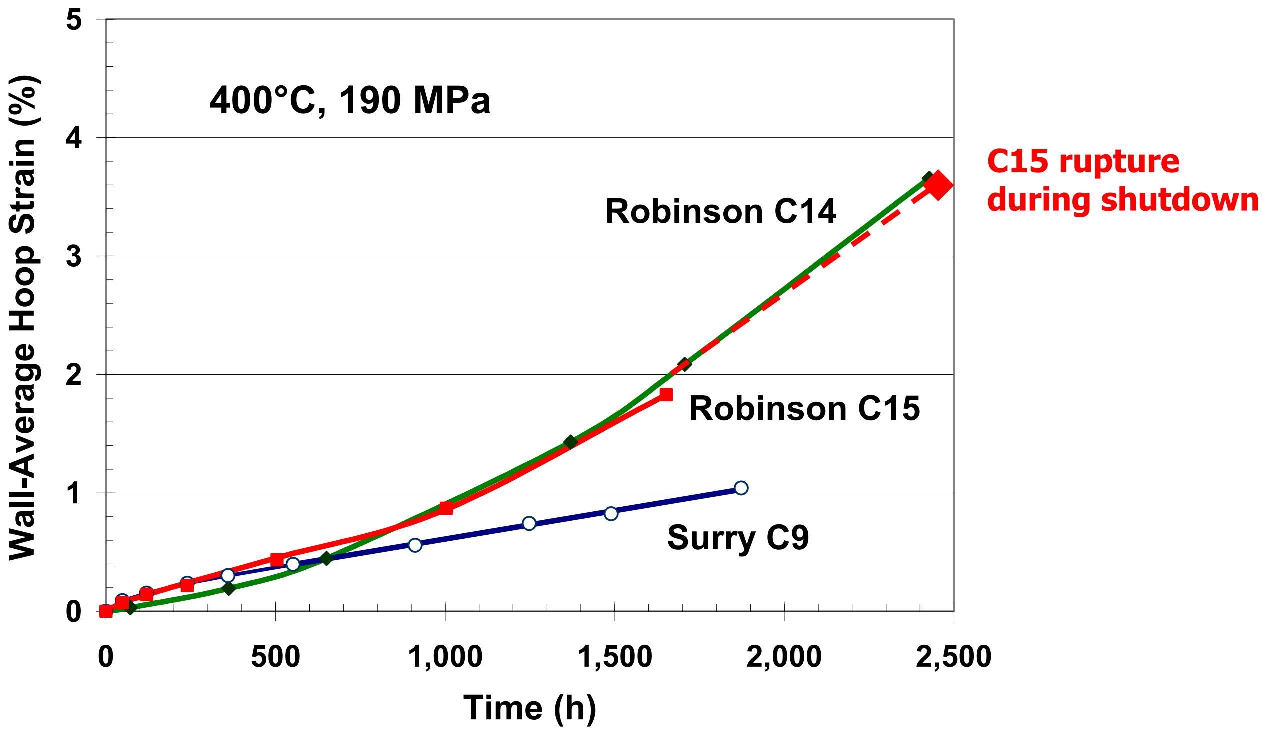
Robinson C14 Sample shows good creep ductility: >3.6 % at 400°C and 190 MPa.

- **Deformation still azimuthally uniform at end of test**
- **Additional creep life likely**



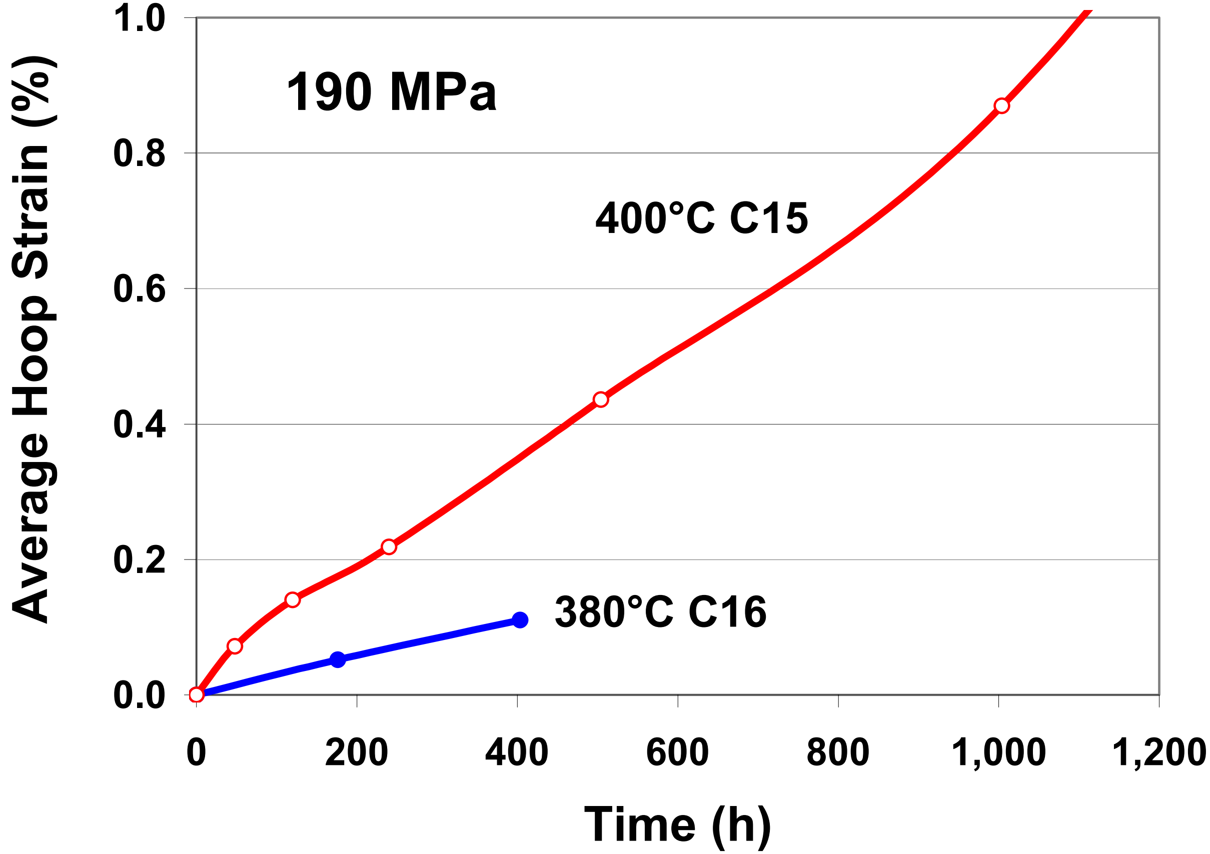
Thermal Creep Tests – H. B. Robinson

- C15 developed a rupture during the final shutdown, which involved cooling from 400°C under full pressure to yield hydride reorientation data.
- Note, 190 MPa is a significant overtest for PWR rods.



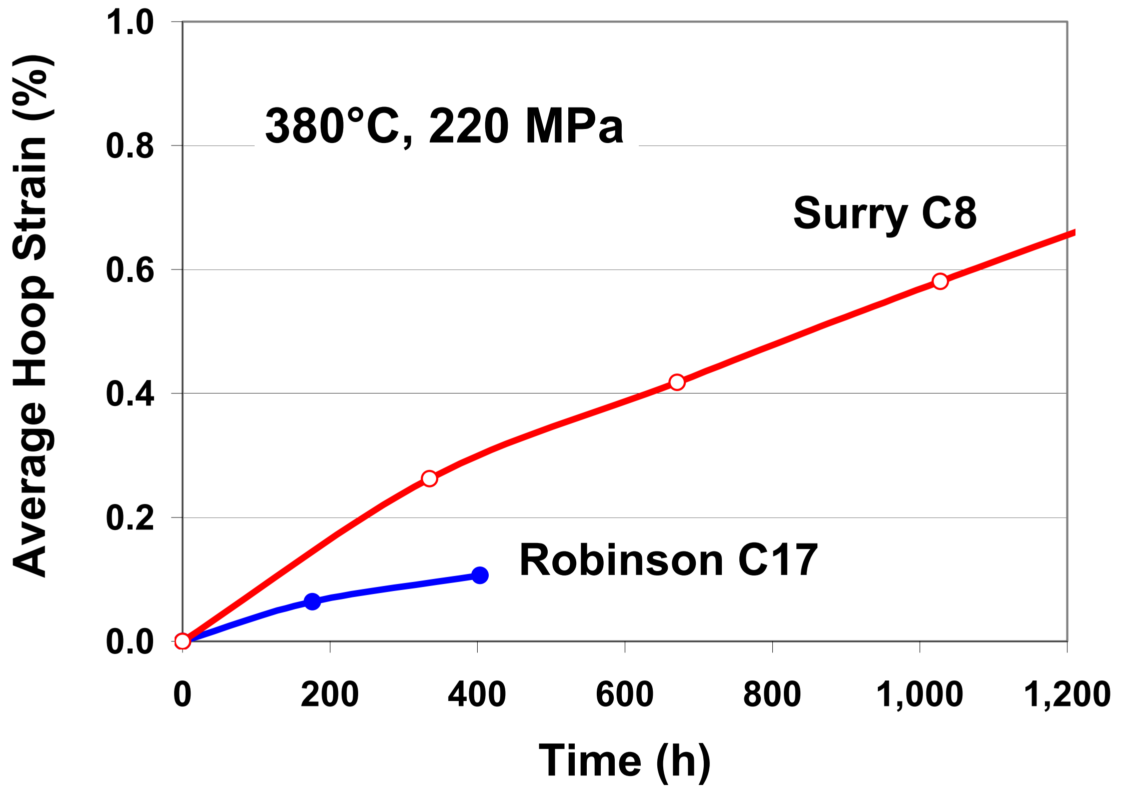
Thermal Creep Tests – H. B. Robinson

- **Temperature Dependency**



Thermal Creep Tests – H. B. Robinson

- Creep rate of H. B. Robinson appears to be smaller than that of Surry at the lower temperature of 380°C.
  - Less recovery at the lower temperature?



Summary and Conclusions

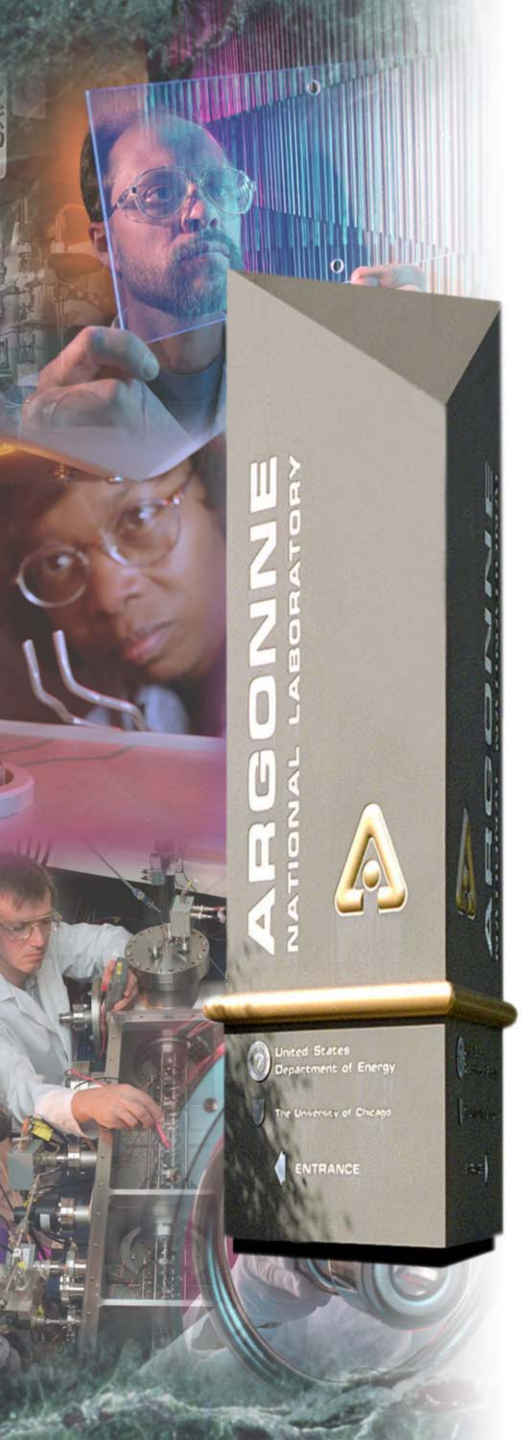
- 15-y storage (with extensive thermal benchmark tests) caused no discernible degradation of the Surry rods.
  - Data useful for dry-cask license extension.
- Significant residual creep ductility has been demonstrated for the post-storage Surry cladding.
  - Findings support NRC ISG-11 (Rev. 2).
- Steady-state creep rates of Surry cladding show strong temperature and stress dependency in the regime tested.
  - Useful for model development and code benchmarking.

Summary and Conclusions (cont'd)

- **Robinson cladding annealing tests showed substantial fraction of radiation hardening can be annealed out at 420-500°C from hours to days.**
- **Early data show high-burnup Robinson cladding possesses good creep ductility and has a creep rate comparable to that of lower-burnup Surry at 400°C.**
  - **Because radiation damage has saturated? Annealing/recovering during tests? Insignificant H effect as long as there is no reorientation?**
  - **More tests are underway.**

Summary and Conclusions *(cont'd)*

- **Unexpected rupture of the H. B. Robinson C15 sample during the final shutdown under pressure requires further investigation**
  - **Was hydride reorientation the cause? If yes, could it happen in real PWR fuel rods? (C15 with full pressure was a significant over-test.)**
- **Hydride reorientation may be a crucial issue for dry-cask storage and transportation, as it can affect cladding integrity. Efforts underway include**
  - **Annealing tests with sealed pressurized samples and with controlled cooling rates.**
  - **Post-creep characterization and mechanical tests.**



Mechanical Property Testing of Fuel Cladding at ANL

R. Daum (rdaum@anl.gov), S. Majumdar, and M. Billone
Energy Technology Division

Nuclear Safety Research Conference, Washington, D.C., October 20-22, 2003

Argonne National Laboratory

Office of Science
U.S. Department of Energy

A U.S. Department of Energy
Office of Science Laboratory
Operated by The University of Chicago

Objective of Mechanical Property Testing

- Determination of stress-strain, deformation, and fracture behavior of Zircaloy-2 and Zircaloy-4 irradiated to highfuel-burnups using ring-stretch, axial, biaxial, bend, and impact specimens relevant to RIA, LOCA, and dry cask storage conditions and transients.
- Develop a database of mechanical properties and limits for inclusion into modeling codes used to analyze high-burnup fuel rods during reactor transients and dry cask storage.

Cladding Type	**Avg. Fast Fluence (E>1 MeV)** **(10^{22} neutrons/cm^2)**	**Rod Avg. Fuel Burnup** (GWd/MTU)	**Max. H Content in Grid Span 4** **(wppm)**
Zircaloy-4	0	0	5-10
Surry Zircaloy-4	0.7	36	310
TMI-1 Zircaloy-4	0.9	50	120
HBR Zircaloy-4	1.4	67	750
Zircaloy-2	0	0	5-10
Limerick Zircaloy-2	1.1	57	70

Key Issues to Consider for High-Burnup Cladding

- **Creep Deformation**
  - *Are rupture strains >1%? ANL data indicates 'yes.'*
  - *Does accumulated creep strain decrease additional plastic ductility?*

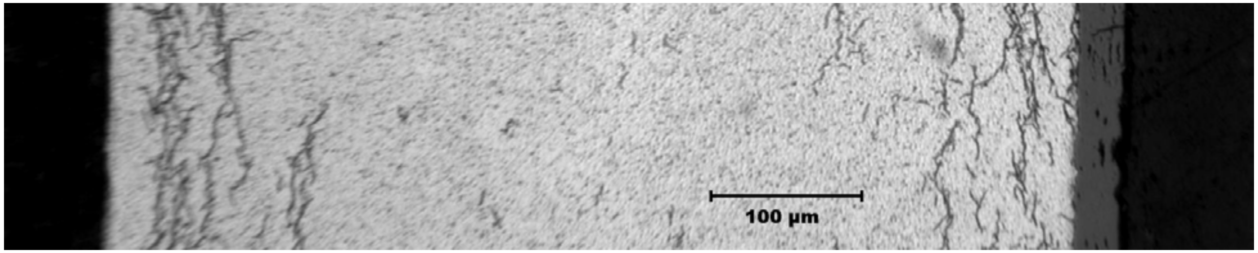
- **Hydrogen Effects**
  - *Do localized hydrides act as crack-initiation sites?*
  - *Does radial reorientation of hydrides increase failure susceptibility?*
  - *Does redistribution of hydrides decrease failure susceptibility?*

- **Accident Loadings**
  - *Which mechanical test? State of Stress? Specimen type?*
  - *Strain rates?*
  - *Temperatures?*

Overview – Focus on Dry Storage Implications

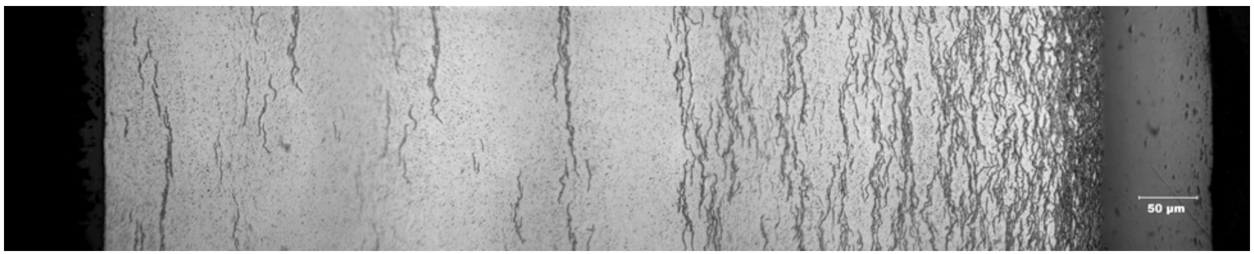
- **Material Characterization (Zircaloy-4)**
- **Mechanical Testing Plans & Procedures**
  - *Testing Plan*
  - *Preparation & Testing Facilities*
- **Evaluation of Testing Techniques**
  - *Descriptions*
  - *Lab-to-Lab Database – An International Perspective*
  - *Relevance to Key Issues*
- **Summary**

Material Characterization (Zircaloy-4)

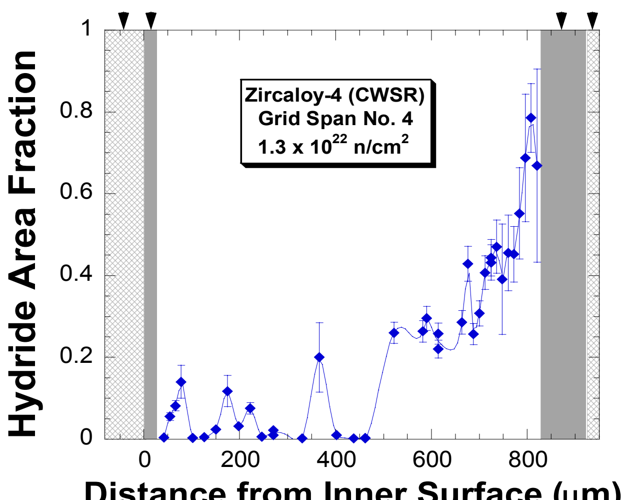


Surry-2
36 GWd/MTU
≈310 wppm hydrogen (max)
15 years in Castor-V/21

Inner Surface **Outer Surface**



H.B. Robinson
67 GWd/MTU
≈750 wppm hydrogen (max)

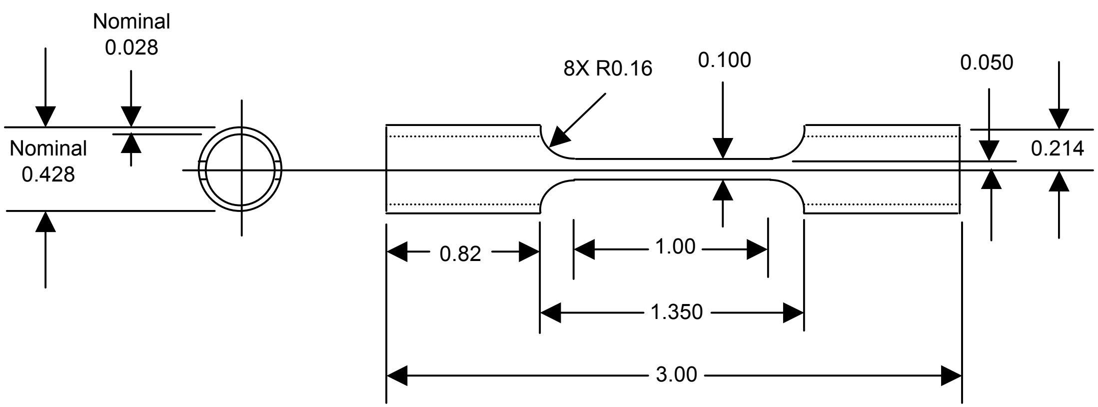


Will redistribution of hydrides occur due to vacuum drying and/or long-term storage?

Will stress-temperature history cause hydride reorientation?
- **» No, Surry-2 after 15 years storage**
- **» Yes, Surry-2 after experiment**

Testing Plan (Zircaloy-4) for Dry Storage

T (°C) \ ε̇	0.1%/s		100%/s	
	Non-irradiated	Irradiated	Non-irradiated	Irradiated
25	FRAM-ANP (7A46)	Surry (1C2) HBR (1C13, TBF)	FRAM-ANP (7A47)	Surry (2C6) HBR (2C6, TBF)
200	FRAM-ANP (7A65)	HBR (TBF)	FRAM-ANP (7A66)	HBR (TBF)
300	FRAM-ANP (7A67)	HBR (2C11)	FRAM-ANP (7A68)	HBR (TBF)
350	FRAM-ANP (7A69)	HBR (TBF)	FRAM-ANP (7A71)	HBR (TBF)
400	FRAM-ANP (7A72)	Surry (2C2) HBR (1C18, TBF)	FRAM-ANP (7A73)	Surry (2C14) HBR (2C8, TBF)



GREEN – Specimen ID (Test Completed)

BLUE – Specimen ID
(Complete & Ready to Test)

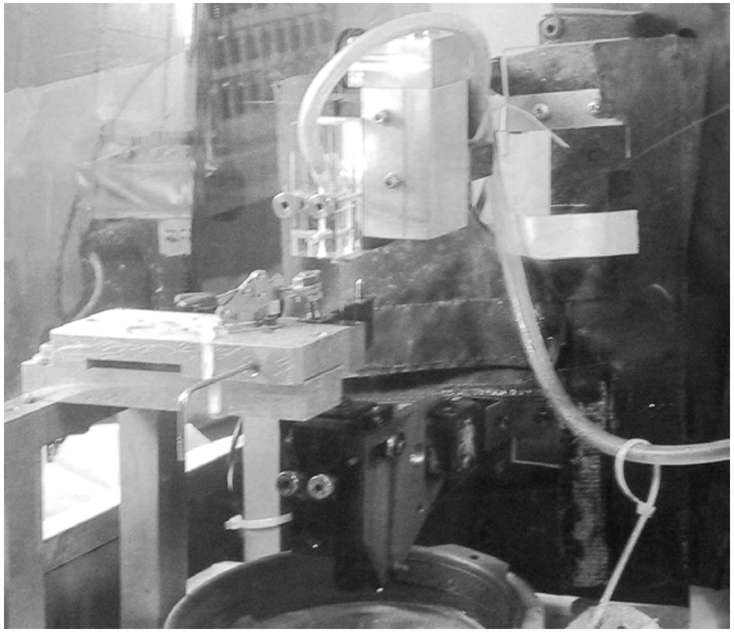
RED – Specimen ID or 'To Be Fabricated
(TBF)' & Tested

Irradiated Specimen Preparation

- **Specimen Inventory:**
  - 6 axial specimens complete & ready for testing (4 – Surry and 2 – HBR)
  - Currently, preparing 3 more HBR axial specimens, along with ring-stretch specimens for LOCA/RIA program

Sectioning	**Completed**
Defueling	**Completed**
Oxide Removal	**Completed**
Endcap Welding	**Completed**
EDM	**Completed**
Glovebox Workup & Furnace Calibration	**On-going**
Measure & Testing	**Not Complete**
Post-test Analysis	**Not Complete**

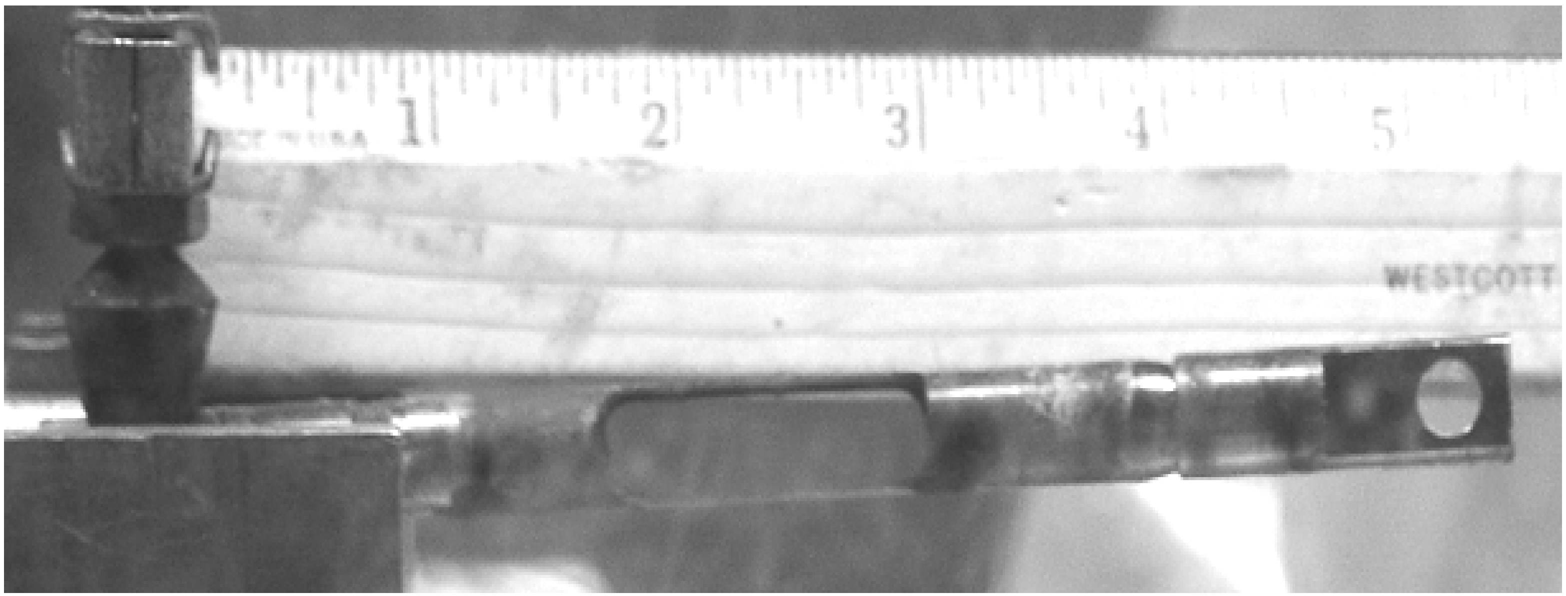
Computer-Controlled, Traveling-Wire Electro-Discharge Machine (EDM) in Hotcell



Irradiated Specimen Preparation

HBR Specimen ID 1C13

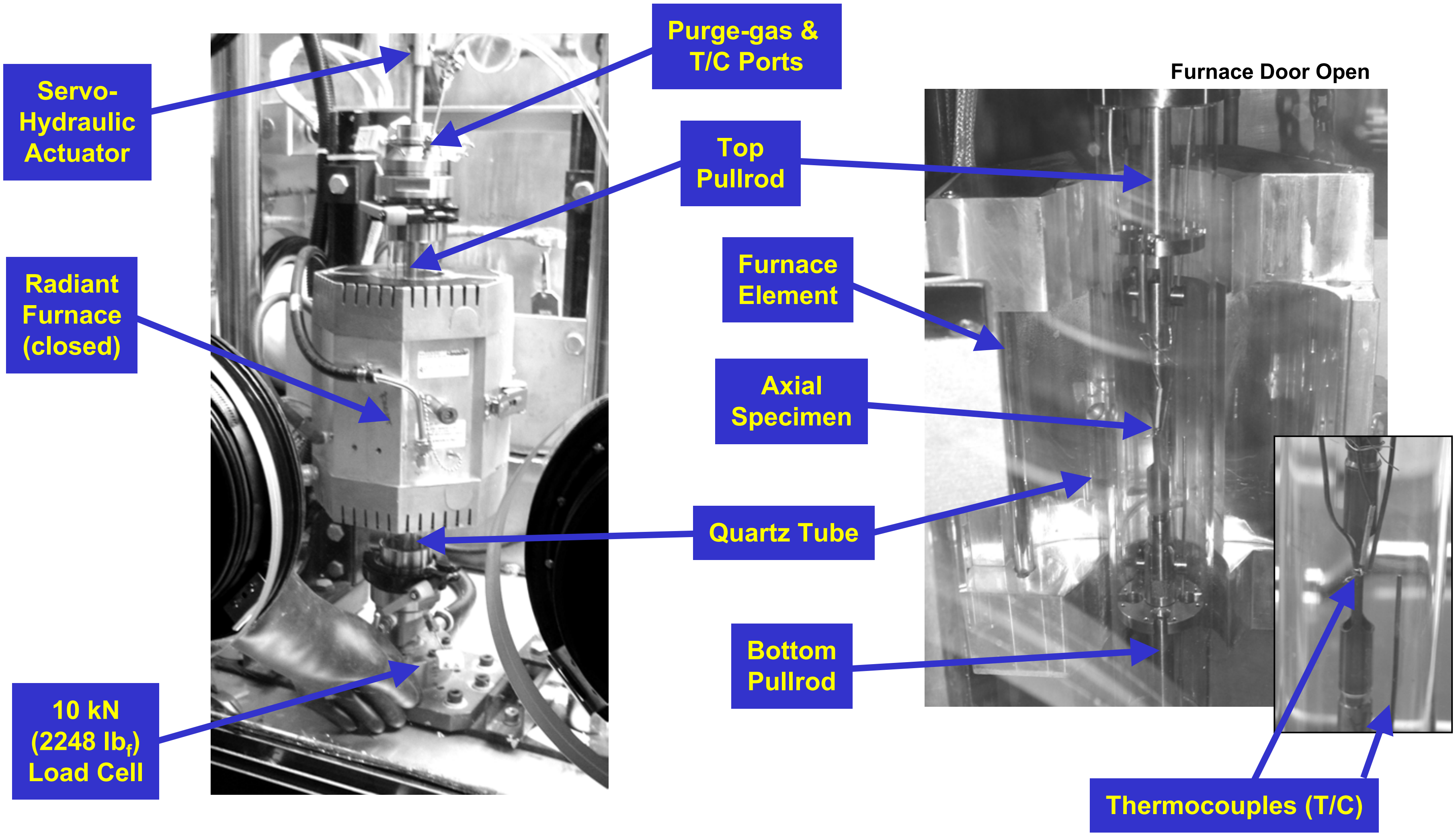
(Ready to Measure & Test)



Testing Facility Upgrades

- **Radiological Glovebox**
  - Primary purpose is contamination control
  - Leaded glass & structure provides γ shielding
  - Experimental Equipment
    - Instron Model 8511 mechanical testing system with infrared furnace
    - Struers Model Duramin-20 imaging and microhardness testing system
  - Awaiting final Design & Safety Verification Reviews

Testing Facility Upgrades



Evaluation of Testing Techniques

- **Pressurized-tube Creep Test – *STEADY-STATE STORAGE***
- **Microhardness - *MECHANICAL PROPERTIES***
- **Uniaxial Tensile Test (z and θ) – *MECHANICAL PROPERTIES***
  - Yield Stress (*YS*), Ultimate Stress (*UTS*), & Uniform Elongation (*UE*)
  - Total elongation (*TE*) is used for comparative purposes
  - Can use results to determine Strain Energy Density (*SED*)
- **Biaxial Tension Test ($1 \leq \theta/z \leq 2$) – *MECHANICAL PROPERTIES***
  - Plane-strain ring specimen for relative ductility, *YS*, *UTS*, & *SED*
  - Tube burst specimen for *YS*, *UTS*, *SED*, & relative ductility
- **Ring Compression Test – *MECHANICAL PROPERTIES***
  - Relative ductility

Evaluation of Testing Techniques (cont'd)

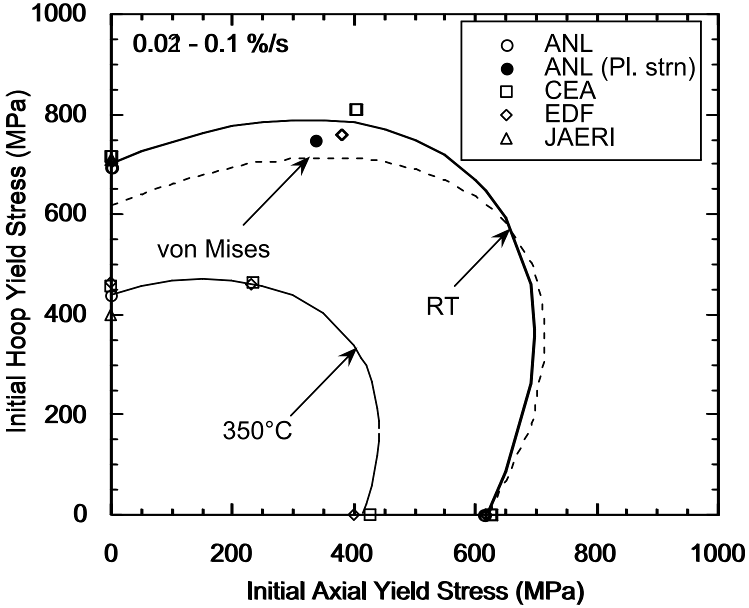
- **3- or 4-Point Bending Test (*UE* < 1%) – *MECHANICAL PROPERTIES*, *HANDLING & ACCIDENT***
  - Fueled pre-creep samples (76-100 mm)
  - Defueled post-creep samples (76-100 mm)
  - Determine effective elastic stiffness for code input
  - Determine failure Bending Moment M_f, axial stress (σ_f) & strain (ε_f), *SED*

- **Impact Tests – *HANDLING & ACCIDENT***
  - High-$\dot{\varepsilon}$ deformation modes
  - Relative impact energy

Evaluation of Testing Techniques (cont'd)

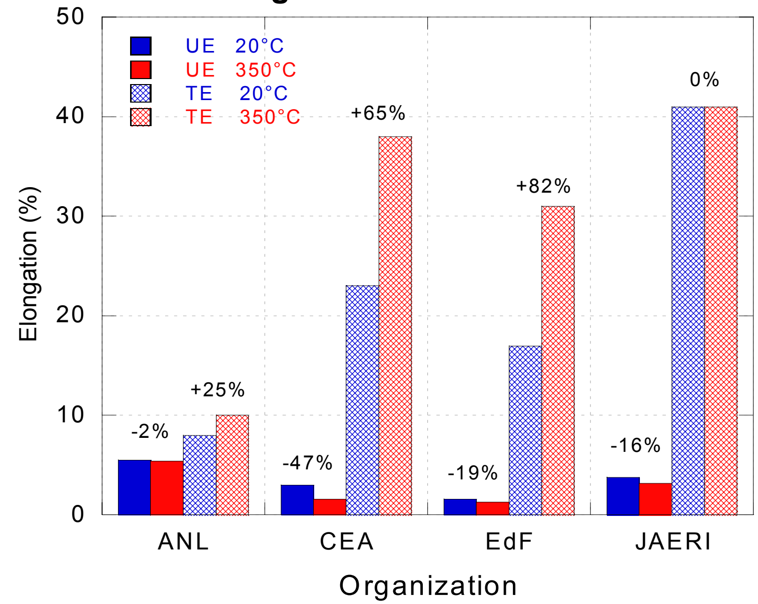
- **Lab-to-Lab Variation (*International Round Robin*) for testing identical material**
  - Differences in specimen design and test procedures

Mechanical Strength – good agreement



Ductility – "relative" agreement

% Change between 20 and 350°C



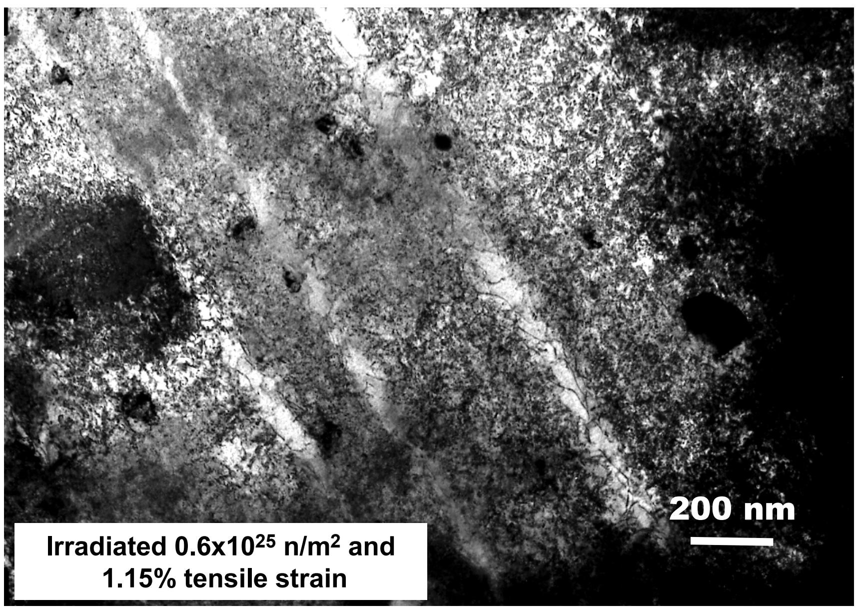
Evaluation of Testing Techniques (cont'd)

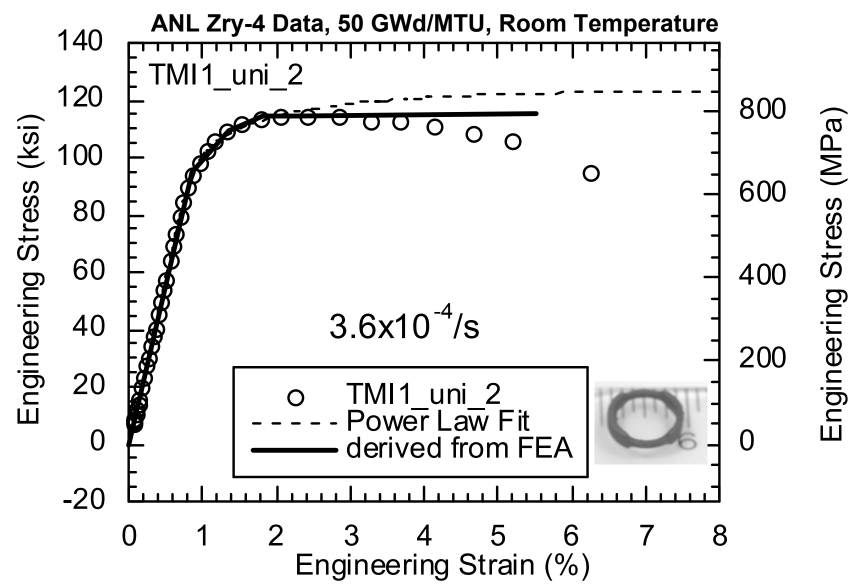
Testing Technique	Primary Relevance	ANL Capable
Pressurized-tube Creep Test	STEADY-STATE STORAGE	yes
Microhardness	*MECHANICAL PROPERTIES*	yes
Axial (*z*) Tensile Test (uniaxial)	*MECHANICAL PROPERTIES*	yes
Hoop (θ) Tensile Test (uniaxial)	*MECHANICAL PROPERTIES*	yes
Plane-strain Tensile Test (biaxial)	*MECHANICAL PROPERTIES*	yes
Tube Burst Tensile Test (biaxial)	*MECHANICAL PROPERTIES*	no
Ring Compression Test	*MECHANICAL PROPERTIES*	yes
3- and 4-point Bend Test	*MECHANICAL PROPERTIES* *HANDLING & ACCIDENT*	yes
Impact Test	*HANDLING & ACCIDENT*	yes

Evaluation of Testing Techniques (cont'd)

- **Relevance to Key Issues – *Creep Deformation***
  - Additional decrease in strain-hardening properties due to increase in creep-induced defect density

C. Regnard, et al., "Activated Slip Systems and Localized Straining Of Irradiated Zr Alloys in Circumferential Loading," Zr in the Nuclear Industry: 13th Inter. Symposium, ASTM STP 1423, pp. 384-399.

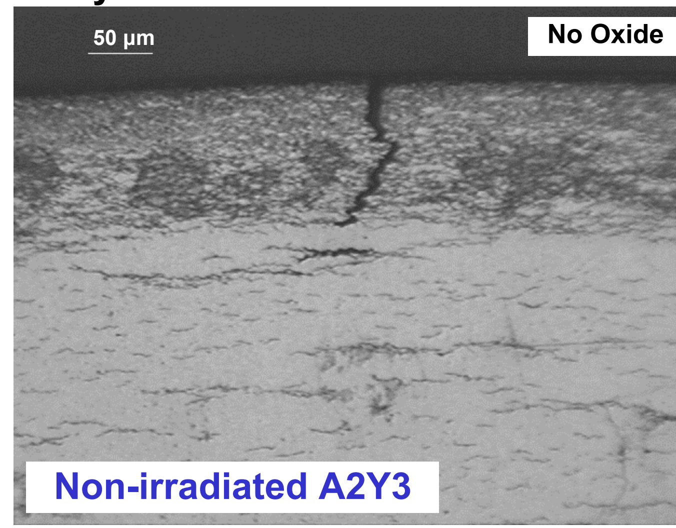


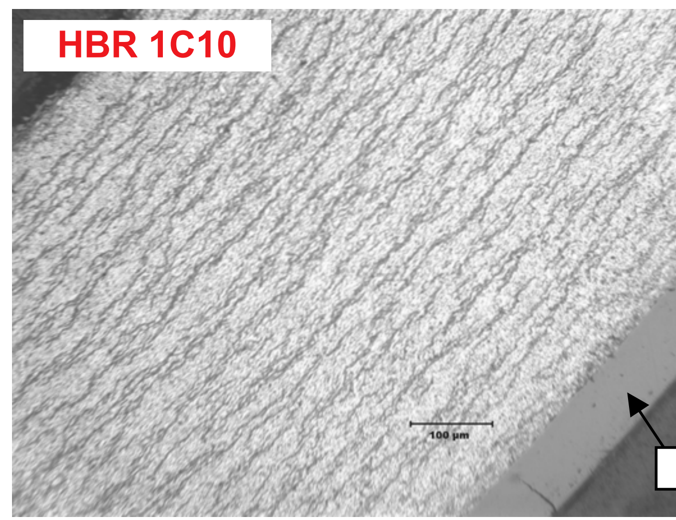


Evaluation of Testing Techniques (cont'd)

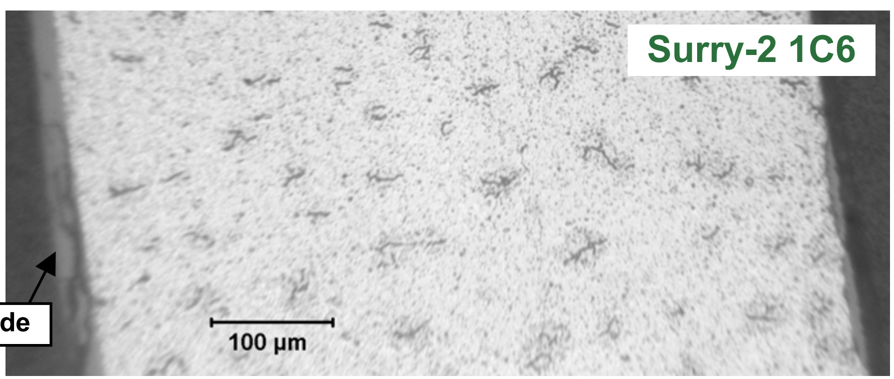
- **Relevance to Key Issues – *Hydrogen Effects***
  - **Hydride layer causes flaw initiation at outer surface**
  - **Re-distribution of hydrides to a more uniform**
  - **Re-orientation of hydrides to a more radial orientation**

Dry Cask License Renewal Criteria





Oxide



Oxide

Evaluation of Testing Techniques (cont'd)

- **Recall Key Issues – *Accident Loadings***
  - **Temperatures 200-400°C per ISG-11**
  - **Strain rates**
    - ***0.1 and 100%/s for MECHANICAL PROPERTIES***
    - ***>100%/s for HANDLING & ACCIDENTS***
  - **Proper selection of pre- and post-creep specimen to represent deformation modes**

F. Yunchang and D.A. Koss, "The Influence of Multiaxial States of Stress on the Hydrogen Embrittlement of Zirconium Alloy Sheet," Metallurgical Transactions A, 16A, April 1985, pp. 675-681.

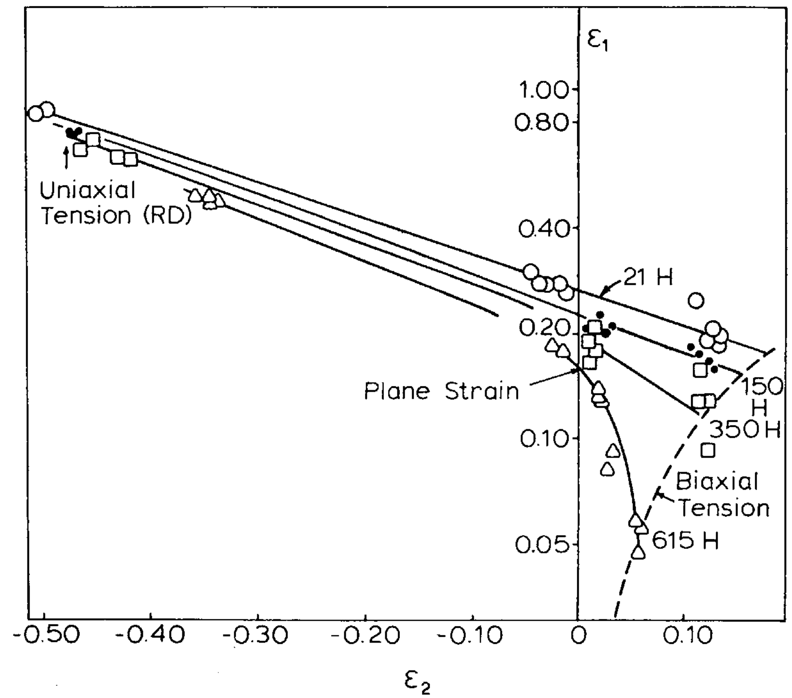


Fig. 3 — A fracture limit diagram for ZIRCALOY-2 sheet at four levels of hydrogen. The major ε_1 and minor ε_2 principal strains in the plane of the sheet at fracture are shown.

Evaluation of Testing Techniques (cont'd)

	Pre-creep Specimens		Post-creep Specimens
Mechanical Properties	Microhardness Axial Tensile Specimen Low strain-rate Bend Specimen *Biaxial Specimen*	↔	Microhardness Axial Tensile Specimen Low strain-rate Bend Specimen *Biaxial Specimen*
		Relative assessment of properties pre- and post-creep	
Handling & Accidents	High strain-rate Bend & Impact	↔	High strain-rate Bend & Impact

Summary

- **Mechanical Properties of Zircaloy-4 cladding from Surry-2 (36 GWd/MTU after 15 years of dry storage) and H.B. Robinson (67 GWd/MTU) for code development and licensing input**
  - Fluence, storage history, and hydrogen effects
  - Initial testing of axial-tensile specimens to commence in Nov. 2003

- **Interpretation of irradiated properties database should be more *relative* than *absolute* to account for lab-to-lab and material-to-material variations.**

- **Pre- and post-creep testing must account for:**
  - Possible, reduction in strain-hardening
  - Hydrogen-induced cracking, hydride re-distribution and re-orientation
  - State of stress relevant to storage handling and accidents